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.

DESIGN THERMAL POWER

1.38 DESIGN THERMAL POWER shall be a design total reactor core heat transfer rate to the reactor coolant of 3411 MWt. See Table 1.3.

ALLOWABLE POWER LEVEL (APL)

1.39 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 exceeded 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, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATIONS (Continued)

Operation with 4 Loops

$K_1 = 1.135$

$K_2 = 0.0130$

$K_3 = 0.000659$

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

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

(ii) For each percent that the magnitude of $(q_t - q_b)$ exceeds -37 percent, the ΔT trip setpoint shall be automatically reduced by 2.3 percent of its value at DESIGN THERMAL POWER.

(iii) For each percent that the magnitude of $(q_t - q_b)$ exceeds +2 percent, the ΔT trip setpoint shall be automatically reduced by 1.8 percent of its value at DESIGN THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

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

where:

ΔT_o	=	Extrapolated ΔT at DESIGN THERMAL POWER
T	=	Average temperature, ^oF
T''	=	Indicated T_{avg} at DESIGN THERMAL POWER 577.1^oF
K_4	=	1.089
K_5	=	$0.0177/^oF$ for increasing average temperature and 0 for decreasing average temperature
K_6	=	0.0011 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 (ΔI) as defined in Note 1 above.

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2.5 percent ΔT span.

Note 4: The channel's maximum trip point shall not exceed its computed trip point by more than 3.4 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 greater than or equal to 1.60% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN less than 1.60% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 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 greater than or equal to 1.60% $\Delta k/k$:

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 3.1.3.5.

c. When in MODE 2##, at least once during control rod withdrawal and at least once per hour thereafter until the reactor is critical.

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 3.1.3.5.

*See Special Test Exception 3.10.1

#With K_{eff} greater than or equal to 1.0

##With K_{eff} less than 1.0

SURVEILLANCE REQUIREMENTS (Continued)

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 coolant 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 ± 1% Δk/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.

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

a. In MODE 4:

1. Greater than or equal to 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. Greater than or equal to 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 greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

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

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 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.

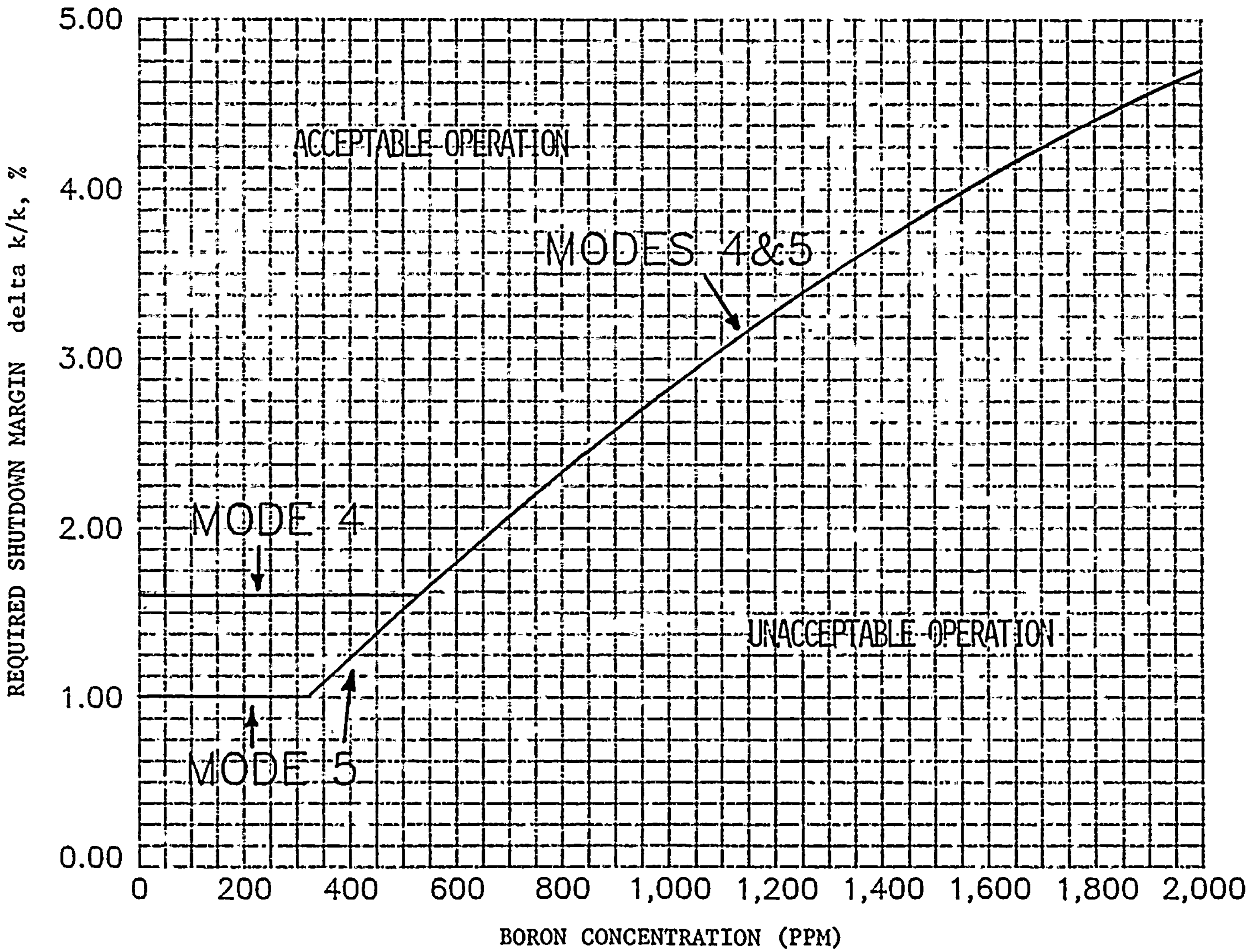


Figure 3.1-3 REQUIRED SHUTDOWN MARGIN

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be greater than or equal to 2000 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 less than 2000 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 greater than or equal to 2000 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 greater than or equal to 2000 gpm through the reactor coolant system.

*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 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the region of acceptable operation in Figure 3.1-2, and

b. Less negative than -3.5×10^{-4} $\Delta k/k/^{o}F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

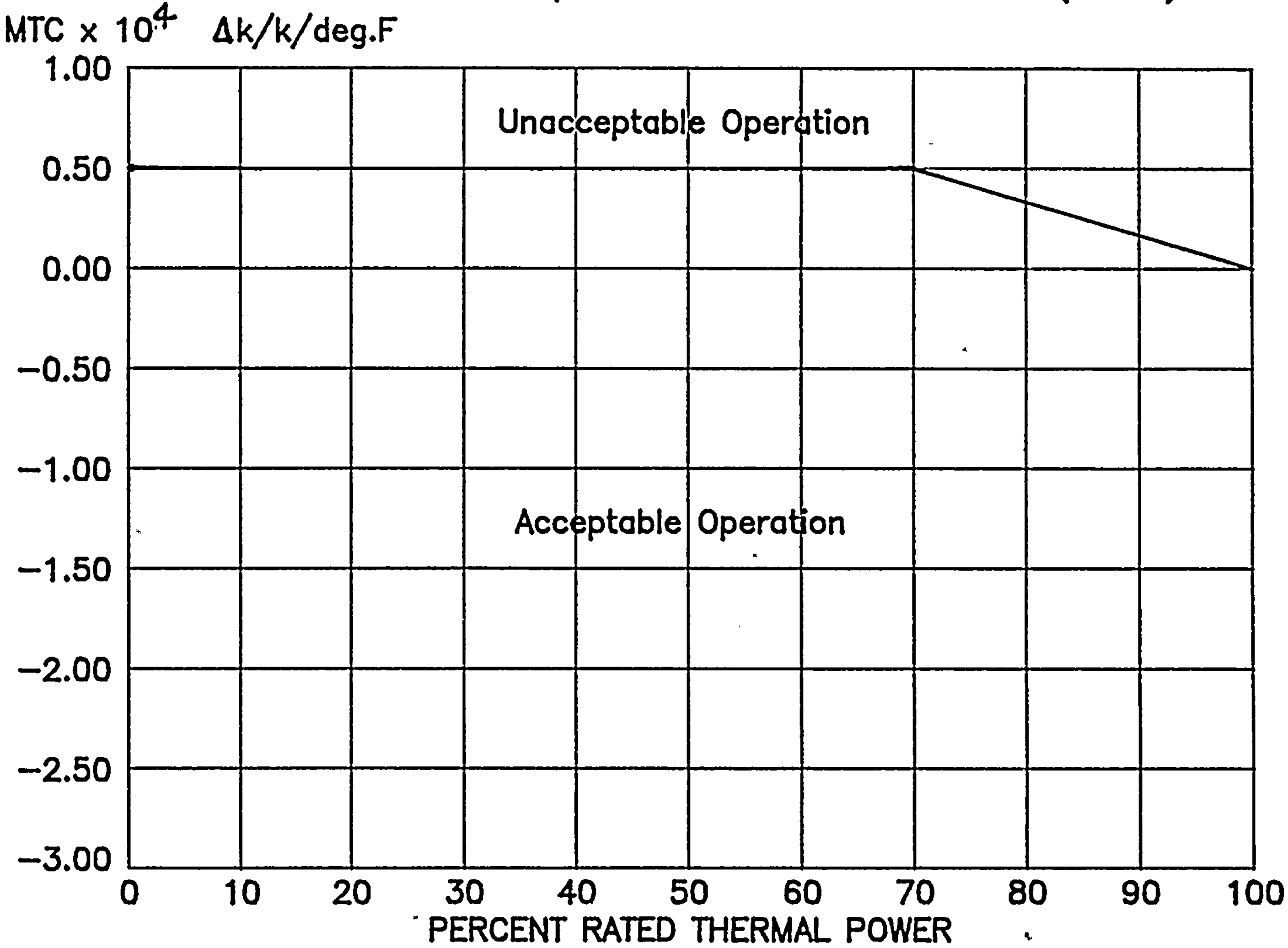
b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

* With K_{eff} greater than or equal to 1.0

See Special Test Exception 3.10.4

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

* 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.

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 170°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 greater than or equal to 2390 psig when tested pursuant to Specification 4.0.5 at least once per 31 days.

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 170°F.

* 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.

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 At least the above required boric acid transfer pump shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 110 psig,

c. Verifying pump operation for at least 15 minutes, and

d. Verifying that the pump is aligned to receive electrical power from an OPERABLE emergency bus.

* 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 usable borated water volume of 4300 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 usable borated water volume of 90,000 gallons,
   2. A minimum boron concentration of 2400 ppm, and
   3. A minimum solution temperature of 80°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 water level volume of the tank, 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.

* 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 - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

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

1. A minimum usable borated water volume of 5650 gallons,

2. Between 20,000 and 22,500 ppm of boron, and

3. A minimum solution temperature of $145^{\circ}F$.

b. The refueling water storage tank with:

1. A minimum contained volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of $80^{\circ}F$.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at $200^{\circ}F$; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the water level of each water source, and

   3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The affected rod is restored to OPERABLE status within the above alignment requirements, or

   2. The affected rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions, and

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

*See Special Test Exceptions 3.10.2 and 3.10.4

LIMITING CONDITION FOR OPERATION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip stepoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE
FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (228 steps) shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°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

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to entering MODE 2:

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

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

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be fully withdrawn (228 steps).

APPLICABILITY: MODES 1* and 2*#

ACTION:

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

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.4
#With K_{eff} greater than or equal to 1.0

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 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 figure, or

c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 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.

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* See Special Test Exceptions 3.10.2 and 3.10.4

With K_{eff} greater than or equal to 1.0.

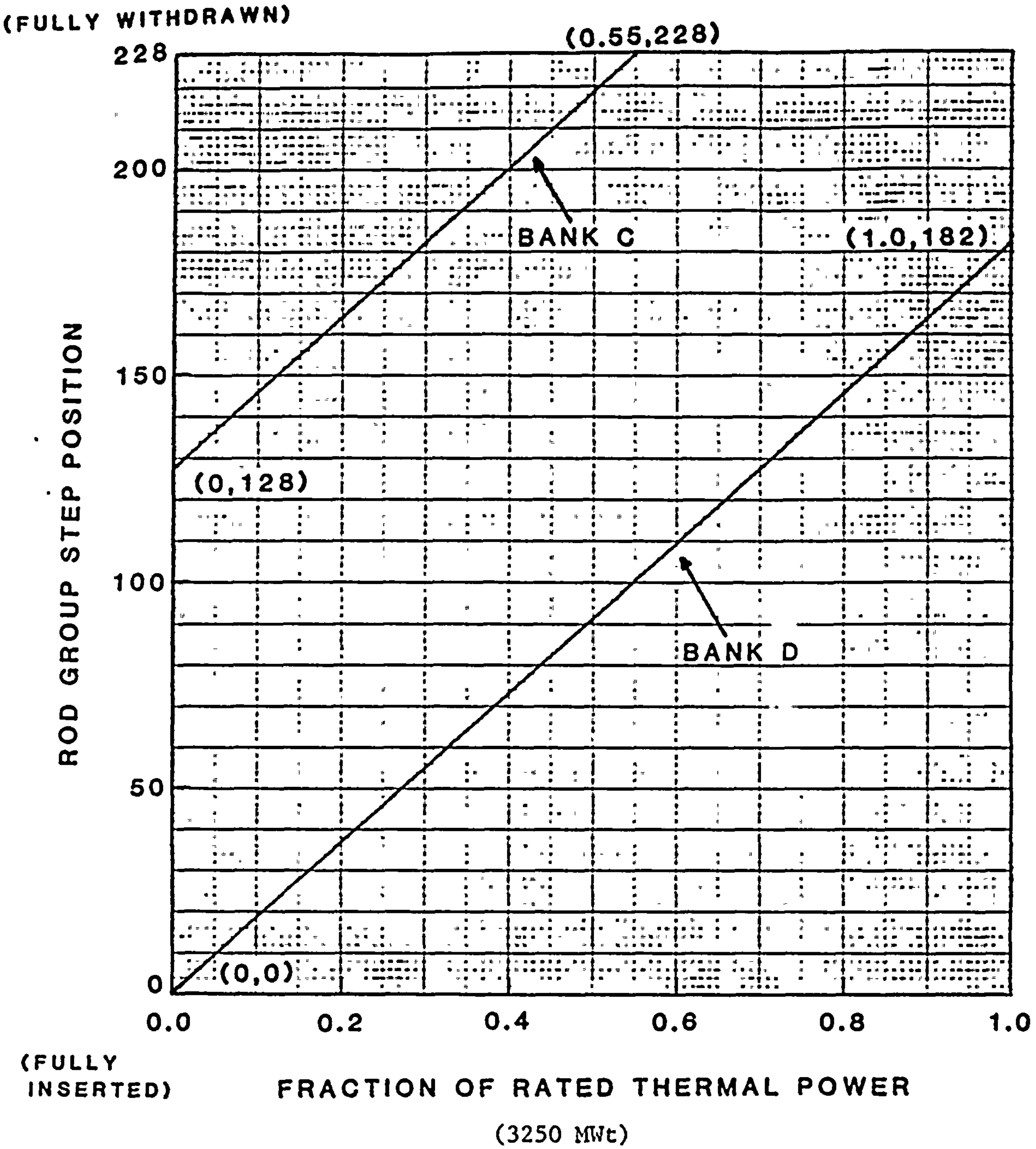

FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER 4 LOOP OPERATION

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AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (±5% or ±3% flux difference units) about a target flux difference.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

   1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

      a) Either restore the indicated AFD to within the target band limits, or

      b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

   2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

      a) POWER OPERATION may continue provided:

         1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

         2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes 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) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (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.

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 and 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 preceding 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:

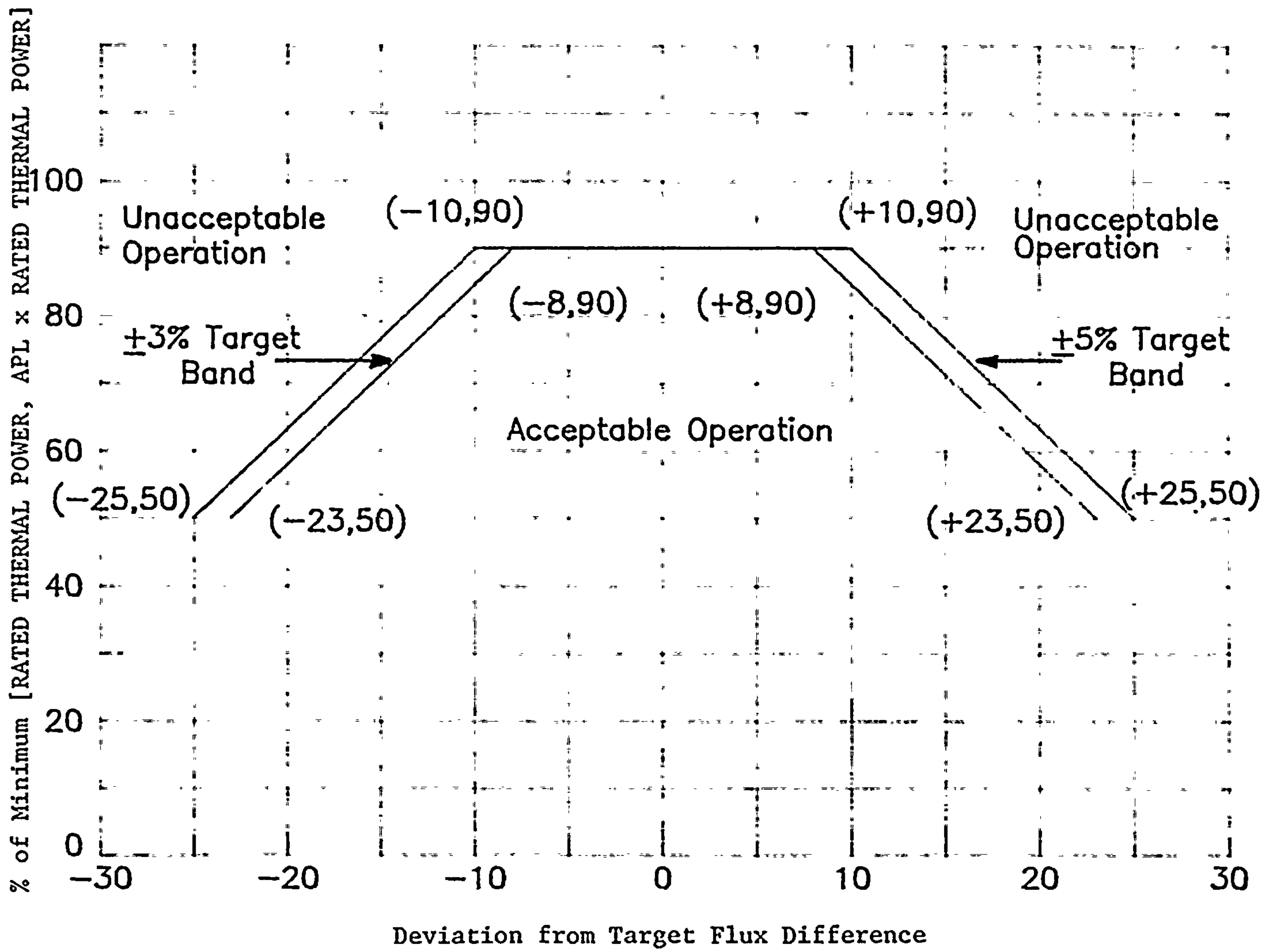
a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

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 of each OPERABLE excore channel 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.

FIGURE 3.2-1 ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE



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	
$F_Q(Z) \le \frac{[2.10]}{P} [K(Z)]$	$F_Q(Z) \le \frac{[2.04]}{P} [K(Z)]$	$P > 0.5$
$F_Q(Z) \le [4.20] [K(Z)]$	$F_Q(Z) \le [4.08] [K(Z)]$	$P \le 0.5$

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

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

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.

FIGURE 3.2-2 EXXON FUEL
K(Z) NORMALIZED VS. CORE HEIGHT

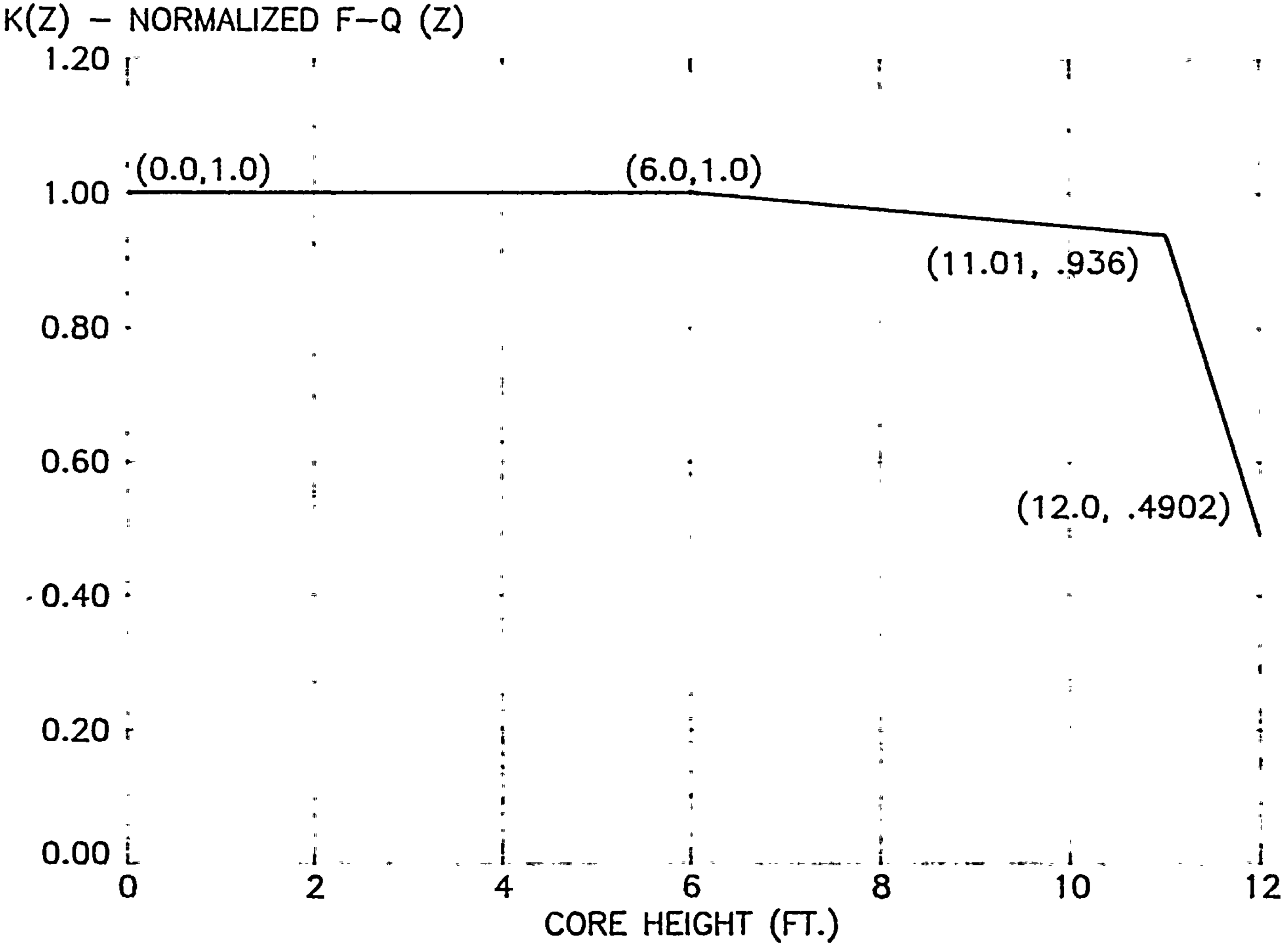
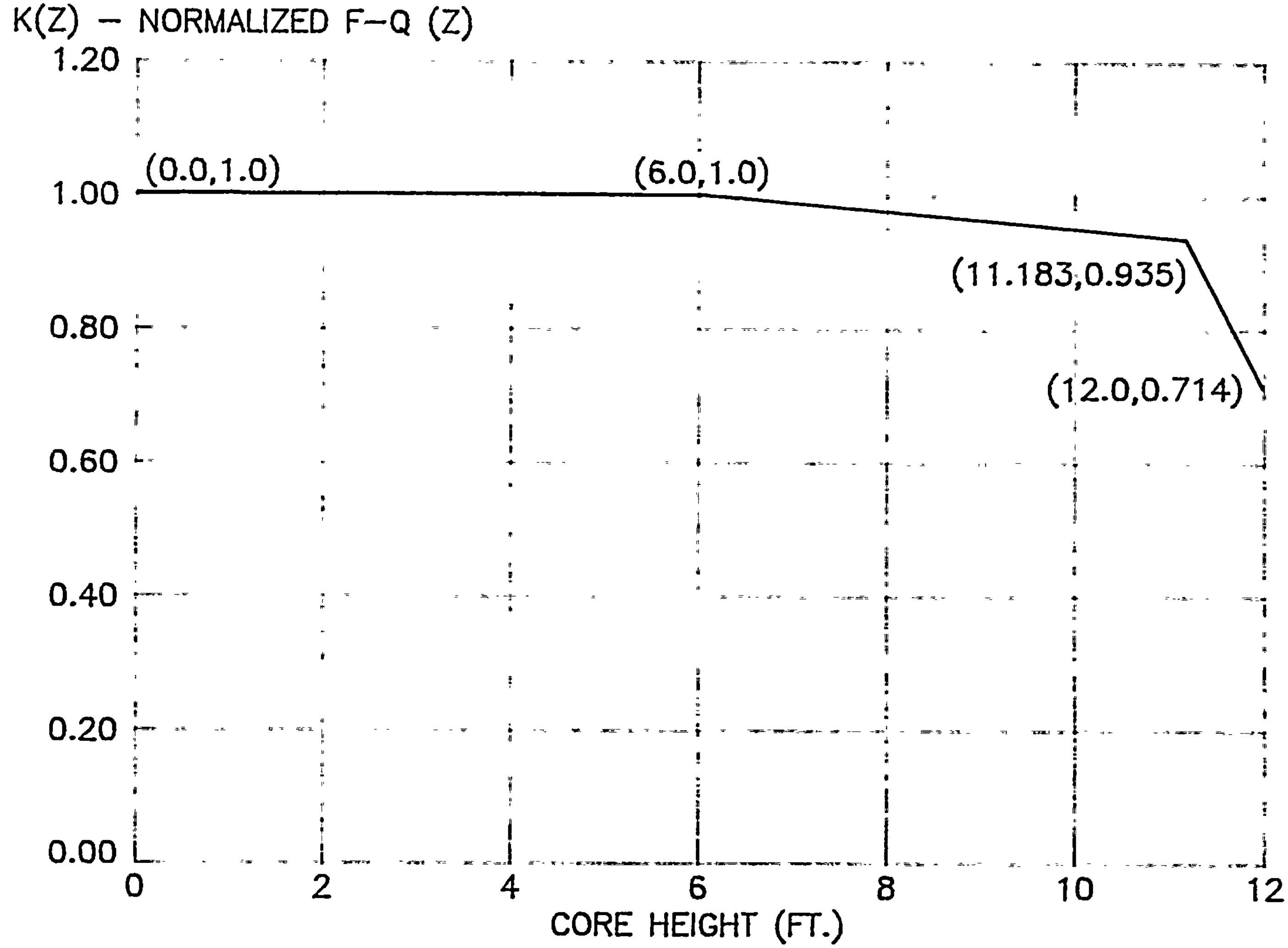


FIGURE 3.2-3 WESTINGHOUSE FUEL
K(Z) NORMALIZED VS. CORE HEIGHT



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

LIMITING CONDITION FOR OPERATION

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

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

and $F^N_{\Delta H} \leq 1.45\ [1 + 0.2\ (1-P)]$ (for Exxon Nuclear Co. fuel)

where P is the fraction of RATED THERMAL POWER

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding 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.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

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

   1. Within 2 hours:

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

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

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

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

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

   1. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

   2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

1. 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.

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. Using the movable incore detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is greater than 75 percent of RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}

b. Pressurizer Pressure

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 percent of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.5.4 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 at RATED THERMAL POWER
Reactor Coolant System T_{avg}	$\leq$ 570.4°F*
Pressurizer Pressure	$\geq$ 2205 psig**
Reactor Coolant System Total Flow Rate	$\geq$ 138.6 x 10^6 lbs/hr***

* Indicated average of at least three OPERABLE instrument loops.

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

***Indicated value.

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

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

Westinghouse Fuel

$$APL = \text{min over Z of } \frac{2.10\ K(Z)}{F_Q(Z) x V(Z) x F_p} \text{ x } 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

Exxon Nuclear Co. Fuel

$$APL = \text{min over Z of } \frac{2.04\ K(Z)}{F_Q(Z) x V(Z) x F_p} \text{ x } 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

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

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

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

  $F_p = 1.02$, or

  $F_p = 1.00$ provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until 2 successive maps indicate that the max over Z of

  $\frac{F_Q(Z)}{K(Z)}$ is not increasing.

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

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

APPLICABILITY: MODE 1

LIMITING CONDITION FOR OPERATION (Continued)

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.

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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 Positive 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
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

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	4	1	7#
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 and below P-8	1/breaker	2	1/breaker per operating loop	1	11#
21. Reactor Trip Breakers	2	1	2	1, 2 3*,4*,5*	1, 13 14
22. Automatic Trip Logic	2	1	2	1, 2 3*,4*,5*	1 14

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 Minimum 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 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 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 of the other channels per Specification 4.3.1.1.1.

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to less than or equal to 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.

ACTION 13 - With one of the diverse trip features (Undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 1. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

REACTOR TRIP SYSTEM INTERLOCKS

DESIGNATION	CONDITION AND SETPOINT	FUNCTION
P-6	With 2 of 2 Intermediate Range Neutron Flux Channels less than $6x10^{-11}$ 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 greater than or equal to 11% of RATED THERMAL POWER or 1 of 2 Turbine First Stage Pressure channels greater than or equal to 37 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 greater than or equal to 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 less than 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 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)(10)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1,2,3*,4*,5*
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	S	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

+ The provisions of Specification 4.0.6 are applicable.

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow-Two Loops	S	R(8)	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R^+	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R^+	M	1, 2
16. Undervoltage-Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency-Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker				
A. Shunt Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1,2,3*,4*,5*
23. Reactor Trip Bypass Breaker	N.A.	N.A.	M(12) and S/U(1)(13)	1,2,3*,4*,5*

$^+$The provisions of Specification 4.0.6 are applicable.

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 absoluted difference greater than 2 percent.

(3) - Compare incore to excore axial imbalance above 15% of RATED THERMAL POWER. Recalibrate if absolute difference greater than or equal to 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)

(10) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(12) - Local manual shunt trip prior to placing breaker in service.

(13) - Automatic Undervoltage Trip.

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 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 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 EITHER					
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*
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

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
OR, COINCIDENT WITH					
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
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	3	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
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

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
COINCIDENT WITH EITHER 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^{*}
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
OR, COINCIDENT WITH					
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	14*

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be bypassed in this MODE below P-11.

##Trip function may be bypassed in this MODE below P-12.

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

####Manually trip all bistables which would be automatically tripped in the event pressure in the associated active loop were less than the pressure in the inactive loop. For example, if loop 1 is the inactive loop then the bistables which indicate low pressure in loops 2, 3, and 4 relative to loop 1 should be tripped.

*The provisions of Specification 3.0.4 are not applicable.

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, operations may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable 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 greater than or equal to 1915 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 less than or equal to Setpoint. Setpoint greater than or equal to 541^{o}F.	P-12 allows the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T_{avg}. P-12 in coincidence with high steam flow will result in a steam line isolation. P-12 affects steam dump blocks. With 3 of 4 T_{avg} channels above the reset value, the manual block of safety injection from high steam flow coincident with either low steam line pressure or low-low T_{avg} is prevented or defeated.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	≤ 13.0#/23.0##
b. Reactor Trip (from SI)	≤ 3.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Auxiliary Feedwater Pumps	Not Applicable
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
d. Containment Air Recirculation Fan	≤ 660.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 11.0
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP, FEEDWATER ISOLATION, AND MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R+	M(3)	1, 2, 3
d. Pressurizer Pressure--Low	S	R+	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R+	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low or Steam Line Pressure--Low	S	R+	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R+	M(3)	1, 2, 3

+ The provisions of Specification 4.0.6 are applicable.

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 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
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 Feed Pumps	N.A.	N.A.	R	1, 2

+The provisions of Specification 4.0.6 are applicable.

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
7. TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3
b. Reactor Coolant Pump Bus Undervoltage	N.A.	R	M	1, 2, 3
8. LOSS OF POWER				
a. 4 kv Bus Loss of Voltage	S	R	M	1, 2, 3, 4
b. 4 kv Bus Degraded Voltage	S	R	M	1, 2, 3, 4

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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, c, and d:

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.

d. At least three of the above coolant loops shall be OPERABLE and in operation above P-12. (Refer to Technical Specification 3.3.2.1, Table 3.3-3 for instrumentation requirements.)

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 less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. With less than the number of operating coolant loops required by item d above, restore the required number of coolant loops within 2 hours or lower the reactor coolant system temperature below P-12.

LIMITING CONDITION FOR OPERATION (Continued)

d. 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.

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.

* 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.

REACTOR COOLANT SYSTEM

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The 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,*

5. Residual Heat Removal - East,**

6. Residual Heat Removal - West,**

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 reactor 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: MODES 4 and 5

ACTION:

a. With less than the above required coolant 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 less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. 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 170°F unless 1) the pressurizer water volume is less than 62% 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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REACTOR COOLANT SYSTEM

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 electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

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

**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 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2485 psig ±1% in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

*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 inoperable 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 of the inoperable PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; restore at least one of the inoperable PORVs 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.b 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.
**The provisions of Specification 4.0.6 are applicable.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 929 and 971 cubic feet,

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

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

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water,

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 80°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

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.

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 Setpoint trip 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 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 4.7-1, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

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TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER		LIFT SETTING (± 1%)*	ORIFICE SIZE
a.	SV-1	1065 psig	16 $in.^2$
b.	SV-1	1065 psig	16 $in.^2$
c.	SV-2	1075 psig	16 $in.^2$
d.	SV-2	1075 psig	16 $in.^2$
e.	SV-3	1085 psig	16 $in.^2$

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

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops an equivalent discharge pressure of greater than or equal to 1240 psig at $60^{\circ}F$ on recirculation flow.

2. Verifying that the steam turbine driven pump develops an equivalent discharge pressure of greater than or equal to 1180 psig at $60^{\circ}F$ and at a flow of greater than or equal to 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:

MODES 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 that is open shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 5 seconds on any closure actuation signal while in HOT STANDBY with T_{avg} greater than or equal to 541°F during each reactor shutdown except that verification of full closure within 5 seconds need not be determined more often than once per 92 days.

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.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. A day tank containing a minimum of 70 gallons of fuel,

2. A fuel storage system containing a minimum of 42,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.6.**

* 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.

**The provisions of Specification 4.0.6 are applicable.

3/4.9 REFUELING OPERATION

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes** and initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

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* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

** 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.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.* The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

* 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.

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 greater than or equal to 2000 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.

SPECIAL TEST EXCEPTIONS

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, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

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

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

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1 and 3.2.4 are suspended, either:

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

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

4.10.2.2 The 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.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the region of acceptable operation shown on Figures 3.4-2 and 3.4-3.

APPLICABILITY: MODE 2

ACTION:

a. With the THERMAL POWER greater than 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figures 3.4-2 and 3.4-3, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figures 3.4-2 and 3.4-3 at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4 and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

NATURAL CIRCULATION TESTS

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.4.1.1 may be suspended during the performance of PHYSICS TESTS and Thermal-Hydraulic Tests, provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5.1 The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during PHYSICS TESTS.

4.10.5.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

PEAKING FACTOR LIMIT REPORT

6.9.1.11 The Peaking Factor Limit Report shall be provided to the NRC Regional Administrator with a copy to Director of Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, containing V(Z) functions for the new cycle at least 15 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In the event that the limit should change, a new or amended Peaking Factor Limit will be submitted 15 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support the content of the Peaking Factor Report will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference Specifications:

a. Inservice Inspection Program Review, Specification 4.4.10.

b. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

c. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

d. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

e. Seismic event analysis, Specification 4.3.3.3.2.

f. Sealed Source leakage in excess of limits, Specification 4.7.7.1.3.

g. Fire Detection Instrumentation, Specification 3.3.3.7.

h. Fire Suppression Systems, Specifications 3.7.9.1, 3.7.9.2, 3.7.9.3 and 3.7.9.4.

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 the 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 with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent confidence that the minimum DNBR for the limiting rod is greater than or equal to the applicable design DNBR limit for each fuel type (as defined below). For 4 loop operation, the improved thermal design procedure is used. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit (as defined below), establishes a design DNBR limit value, which must be met in plant safety analyses, using values of input parameters without uncertainties.

The table below indicates the relationship between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design.

4 Loop Operation

	Westinghouse Fuel (15x15 OFA)		Exxon Nuclear Co. Fuel (15x15)	
	(WRB-1 Correlation)		(W-3 Correlation)	
	Typical Cell*	Thimble Cell**	Typical Cell*	Thimble Cell**
Correlation Limit	1.17	1.17	1.30	1.30
Design Limit DNBR	1.32	1.31	1.58	1.50
Safety Analysis Limit DNBR	1.69	1.69	1.58	1.50

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

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

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 axial power distribution, 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 drops below 90% of nominal full loop flow. Above the P-8 setpoint, less than or equal to 31% of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow.

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 auxiliary 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 less than or equal to 0.71×10^6 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. A 0.1 second time delay is incorporated in each of these trips to prevent spurious reactor trips from momentary electrical power transients.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-7. Each of the turbine trips provide turbine protection and reduce 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.

Safety Injection Input from ESF

If a reactor trip has not already been generated by the reactor protective instrumentation, the ESF automatic actuation logic channels will initiate a reactor trip upon any signal which initiates a safety injection. This trip is provided to protect the core in the event of a LOCA. The ESF instrumentation channels which initiate a safety injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB resulting from the opening of any one pump breaker above P-8 or the opening of two or more pump breakers below P-8. These trips are blocked below P-7. The open/close position trips assure a reactor trip signal is generated before the low flow trip set point is reached. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System.

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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 1.60% $\Delta k/k$ is initially required to control the reactivity transient and automatic ESF is assumed to be available. With T_{avg} less than 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.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 2000 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 2000 GPM will circulate an equivalent Reactor Coolant System volume of 12,612 ± 100 cubic feet in approximately 45 minutes. The reactivity change rate associated with boron reductions will therefore be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC at the beginning, and near the end of each fuel cycle is adequate to confirm the MTC value since this coefficient changes slowly due

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

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.

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 minor 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 170°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.

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from all operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability, usable volume requirement, is 5641 gallons of 20,000 ppm borated water from the boric acid storage tanks or 99,598 gallons of 2400 ppm borated water from the refueling water storage tank. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5.

BORATION SYSTEMS (Continued)

With the RCS average 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.

The boration capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 2890 gallons of 20,000 ppm borated water from the boric acid storage tanks or 76,937 gallons of 2400 ppm borated water from the refueling water storage tank. The boration source volumes of Technical Specification 3.1.2.7 have been conservatively increased to 4300 gallons from the boric acid storage tank and 90,000 gallons from the RWST. These values were chosen to be consistent with Unit 2. The Unit 2 value for the boric acid storage tank volume includes sufficient boric acid to borate to 2000 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis for a rod ejection accident.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} greater than or equal to 541°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reator trip at operating conditions.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to 1.69 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

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

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

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

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

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

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels above 50% of RATED THERMAL POWER. For THERMAL POWER levels below 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less). During operation at THERMAL POWER levels between 15% and 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less), the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

The upper bound limit (90% or 0.9 x APL of RATED THERMAL POWER (whichever is less)) on AXIAL FLUX DIFFERENCE assures that the $F_Q(Z)$ envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The lower bound limit (50% of RATED THERMAL POWER) is based on the fact that at THERMAL POWER levels below 50% of RATED THERMAL POWER, the average linear heat generation rate is half of its nominal operating value and below that value, perturbations in localized flux distributions cannot affect the results of ECCS or DNBR analyses in a manner which would adversely affect the health and safety of the public.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

The bases and methodology for establishing these limits is presented in topical report WCAP - 8385, "Power Distribution Control and Load Following Procedures."

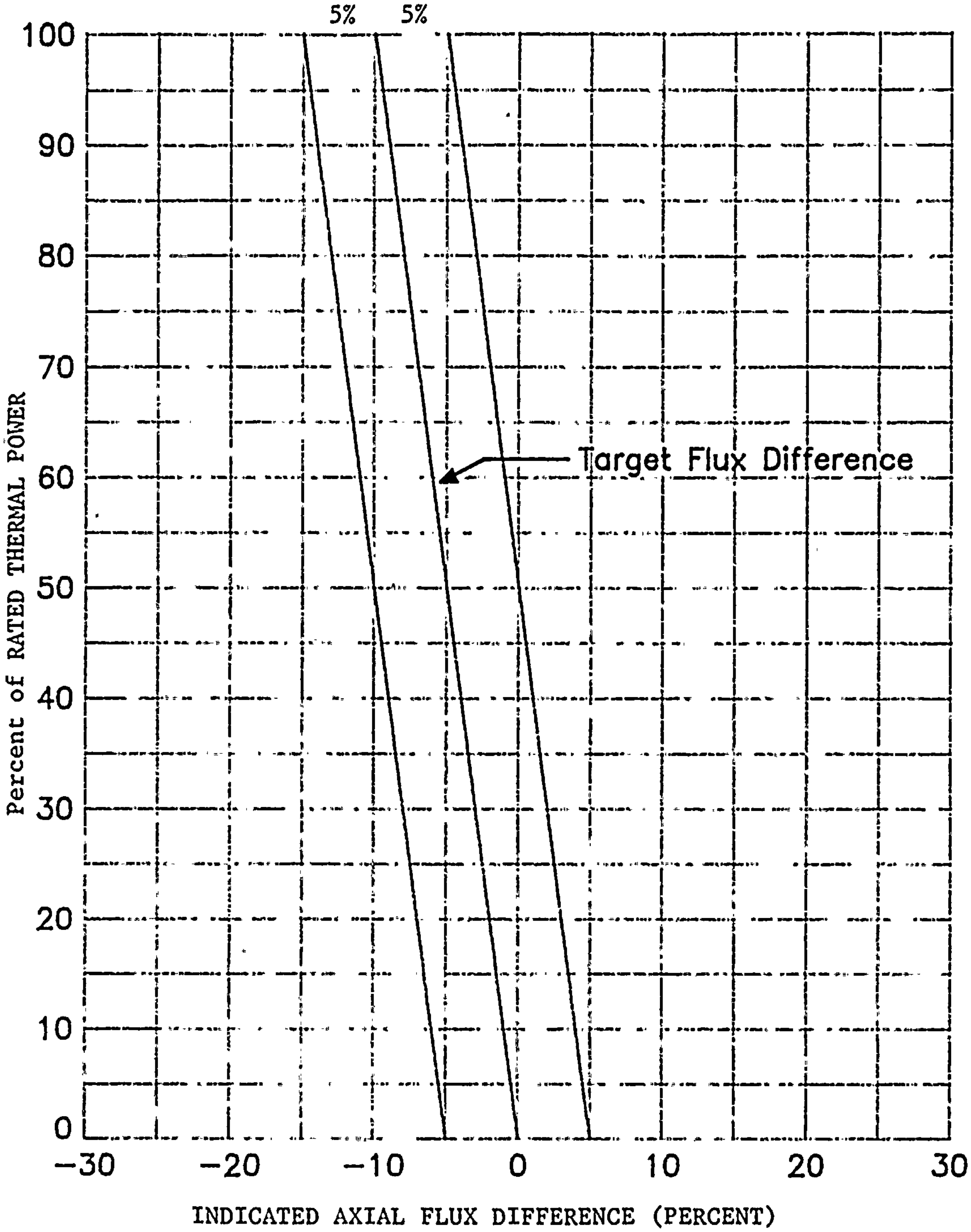

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL

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

The limits on heat flux and nuclear enthalpy rise hot channel factors 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 hot channel factors are 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 insure that the hot channel factor 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 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

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

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits, provided conditions (a) through (d) above are maintained.

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.

When $F^N_{\Delta H}$ is measured, experimental error must be allowed for, and 4% is the appropriate allowance for a full core map taken with the incore detection system. This 4% measurement uncertainty has been included in the design DNBR limit value. The specified limit for $F^N_{\Delta H}$ also contains an additional 4% allowance for uncertainties. The total allowance is based on the following considerations:

a. abnormal perturbations in the radial power shape, such as from rod misalignment, affect $F^N_{\Delta H}$ more directly than F_Q,

b. although rod movement has a direct influence upon limiting F_Q to within its limit, such control is not readily available to limit $F^N_{\Delta H}$, and

c. errors in prediction for control power shape detected during startup PHYSICS TESTS can be compensated for in F_Q, by restricting axial flux distributions. This compensation for $F^N_{\Delta H}$ is less readily available.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

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 for each percent of tilt in excess of 1.0.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated to be adequate to maintain the applicable design limit DNBR values for each fuel type (which are listed in the bases for Section 2.1.1) throughout each analyzed transient. The indicated values of T_{avg} and flow include allowances for instrument errors. Measurement uncertainties have been accounted for in determining the DNB parameters' limit values.

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 12-hour surveillance of the RCS flow measurement. The total flow is measured after each refueling based on a secondary side calorimetric and measurements of primary loop temperature.

3/4.2.6 ALLOWABLE POWER LEVEL - APL

Constant Axial Offset Control (CAOC) operation 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 Peaking Factor Limit Report and applied by the Technical Specifications provides the means for predicting the maximum $F_Q(Z)$ distribution anticipated during operation using CAOC 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 CAOC. This comparison is done by calculating APL, as defined in specification 3.2.6.

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3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.69 during all normal operations and anticipated transients. 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 below P-7 with one or more RCS cold legs less than or equal to 170°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 RCP's to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 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 set point. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accomodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation conditions.

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 a 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.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50. The value of the minimum RWST temperature in Technical Specification 3.5.5 has been conservatively changed to 80°F to increase the consistency between Units 1 and 2. The lower RWST temperature results in lower containment pressure from containment spray and safeguards flow assumed to exit the break. Lower containment pressure results in increased flow resistance of steam exiting the core thereby slowing reflood and increasing PCT.

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 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 approximately 121 percent of the total secondary steam flow of 14,120,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per operable 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 = 1, 2 or 3.

X = Total relieving capacity of all safety valves per steam line = 4,288,450 lbs/hour.

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

(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.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The boron concentration requirement of specification 3.9.1.b has been conservatively increased to 2400 ppm to agree with the minimum concentration of the RWST.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

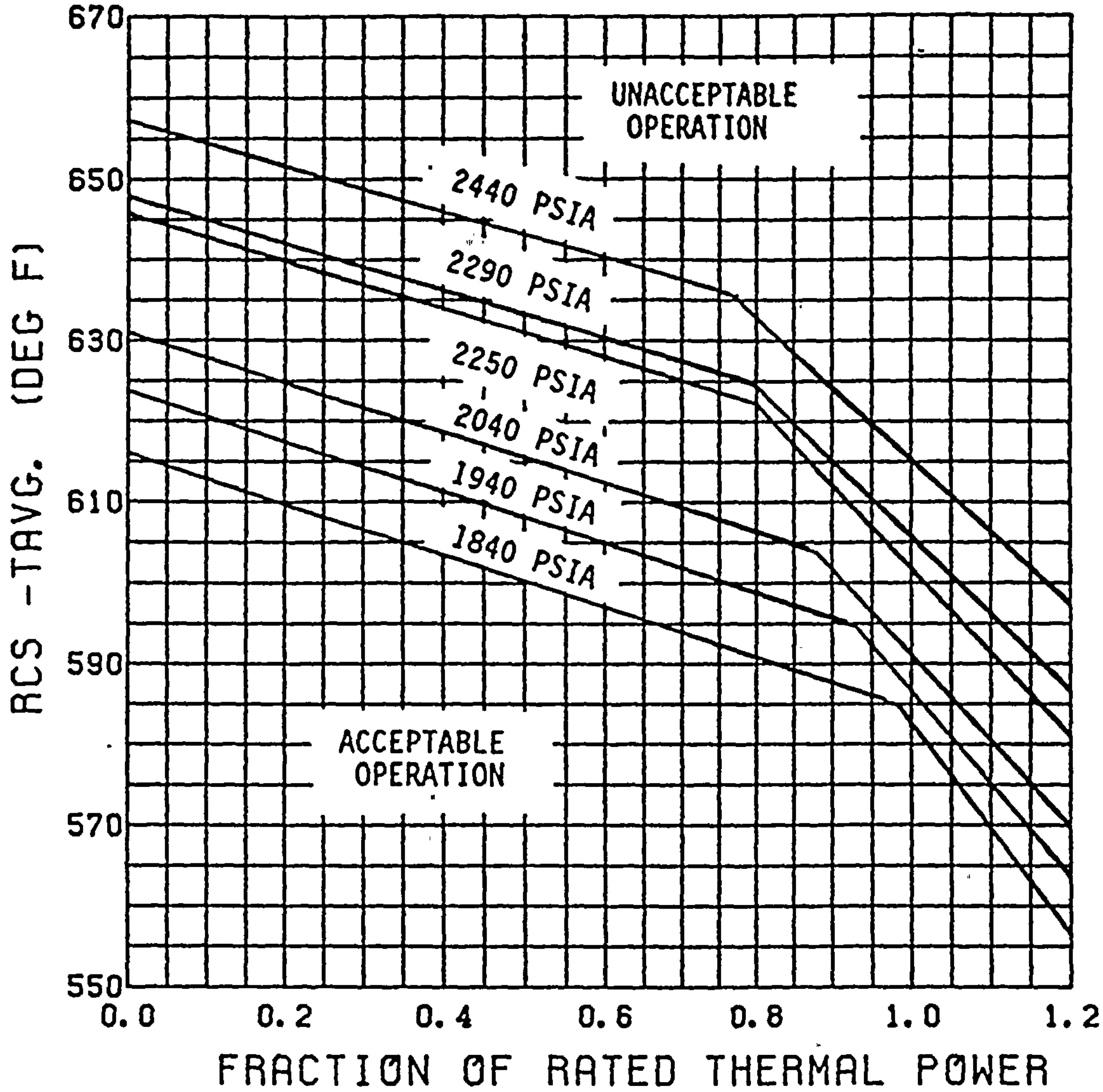
The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T AVG, DEG F)
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)
2250	(0.00,645.9) , (0.80,622.3) , (1.20,580.9)
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

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be:

a. In MODE 4:

1. Greater than or equal to 1.6% Δk/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. Greater than or equal to 1.0% Δk/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 greater than or equal to 10 gpm of 20,000 ppm boric acid solution or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

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

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

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant through the reactor coolant system shall be greater than or equal to 2000 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 less than 2000 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 greater than or equal to 2000 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 greater than or equal to 2000 gpm through the reactor coolant system.

* 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 (MODES 1, 2, 3, and 4) or 3.1.2.7.b.2 (MODES 5 and 6).

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the region of acceptable operation in Figure 3.1-2, and

b. Less negative than -3.9×10^{-4} $\Delta k/k/^{o}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.4.a - MODES 1 and 2* only#
Specification 3.1.1.4.b - MODES 1, 2 and 3 only#

ACTION:

a. With the MTC more positive than the limit of 3.1.1.4.a above:

1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limits within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.

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

b. With the MTC more negative than the limit of 3.1.1.4b above, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.0

See Special Test Exception 3.10.3

SURVEILLANCE REQUIREMENTS

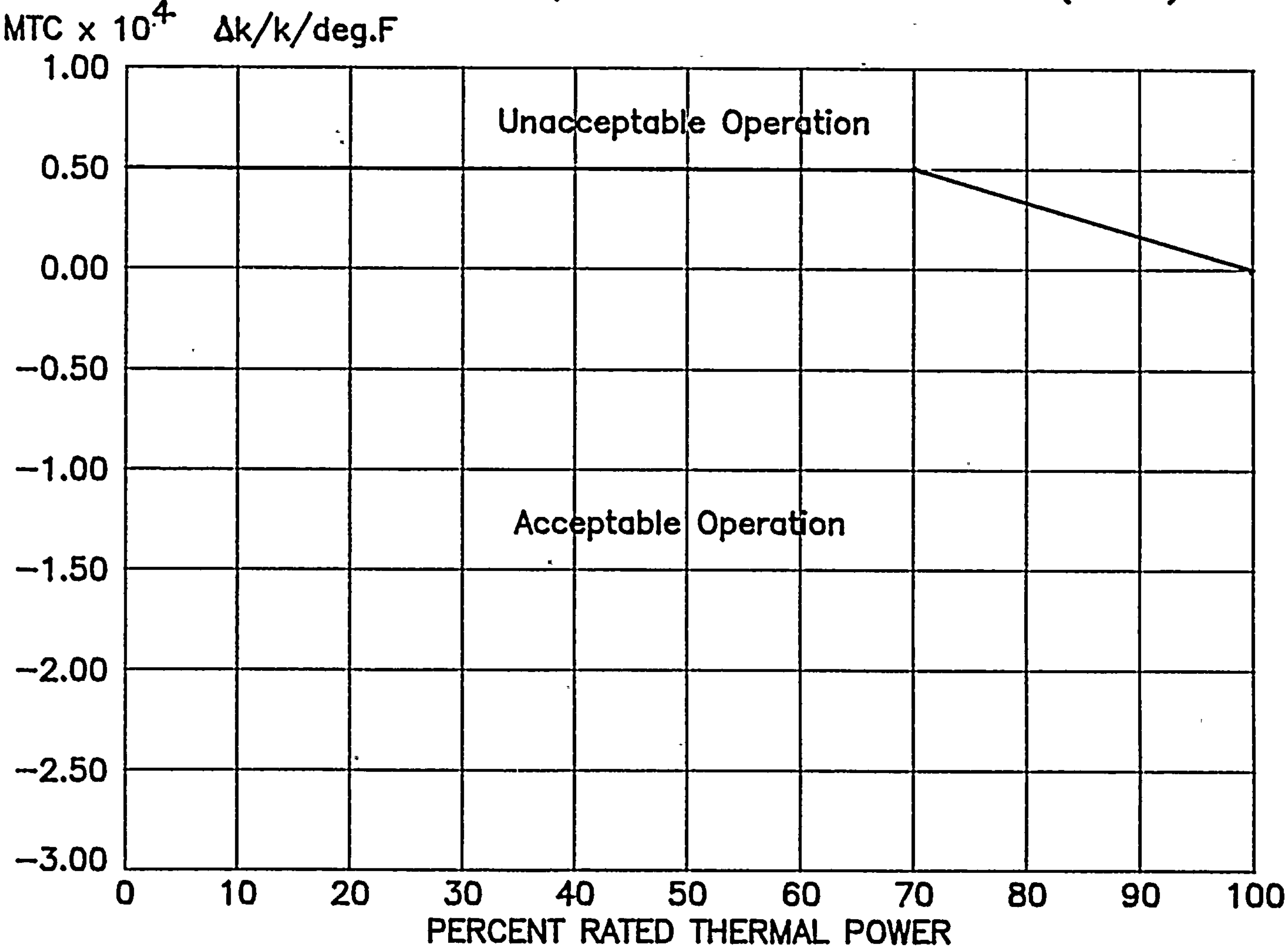
4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.4.a, above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. The MTC shall be measured at any THERMAL POWER and compared to -3.0×10^{-4} $\Delta k/k/^{\circ}F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.0×10^{-4} $\Delta k/k/^{\circ}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.4.b, at least once per 14 EFPD during the remainder of the fuel cycle.

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if only the boric acid storage tank in Specification 3.1.2.7a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if only the refueling water storage tank in Specification 3.1.2.7b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes* until at least one injection path is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 145°F when a flow path from the boric acid tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

* 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.

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 greater than or equal to 2390 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.

* 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 usable borated water volume of 4300 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 usable borated water volume of 90,000 gallons,

2. A minimum boron concentration of 2400 ppm, and

3. A minimum solution temperature of 80°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.

* 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.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

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

1. A minimum usable borated water volume of 5650 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 volume of 350,000 gallons of water,

2. Between 2400 and 2600 ppm of boron, and

3. A minimum solution temperature of 80°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

2. Verifying the contained borated water volume of each water source, and

3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than addressed by ACTION a, above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The affected rod is restored to OPERABLE status within the above alignment requirements, or

   2. The affected rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions, and

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

*See Special Test Exceptions 3.10.2 and 3.10.3

LIMITING CONDITION FOR OPERATION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip stepoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figure 3.1-1; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

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 (228 steps) shall be less than or equal to 2.2 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 541°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 entering MODE 2:

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

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

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn (228 steps).

APPLICABILITY: MODES 1* and 2*#

ACTION:

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

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 12 hours thereafter.

*See Special Test Exceptions 3.10.2 and 3.10.3
#With K_{eff} greater than or equal to 1.0

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 figure, or

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

SURVEILLANCE REQUIREMENTS

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

* See Special Test Exceptions 3.10.2 and 3.10.3

With K_{eff} greater than or equal to 1.0.

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AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band (±5% or ±3% flux difference units) about a target flux difference.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:

1. Above 90% of 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:

   a) Either restore the indicated AFD to within the target band limits, or

   b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.

2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:

   a) POWER OPERATION may continue provided:

      1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

      2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes 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) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limit of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

*See Special Test Exception 3.10.2

FIGURE 3.2-1 ALLOWABLE DEVIATION FROM TARGET FLUX DIFFERENCE

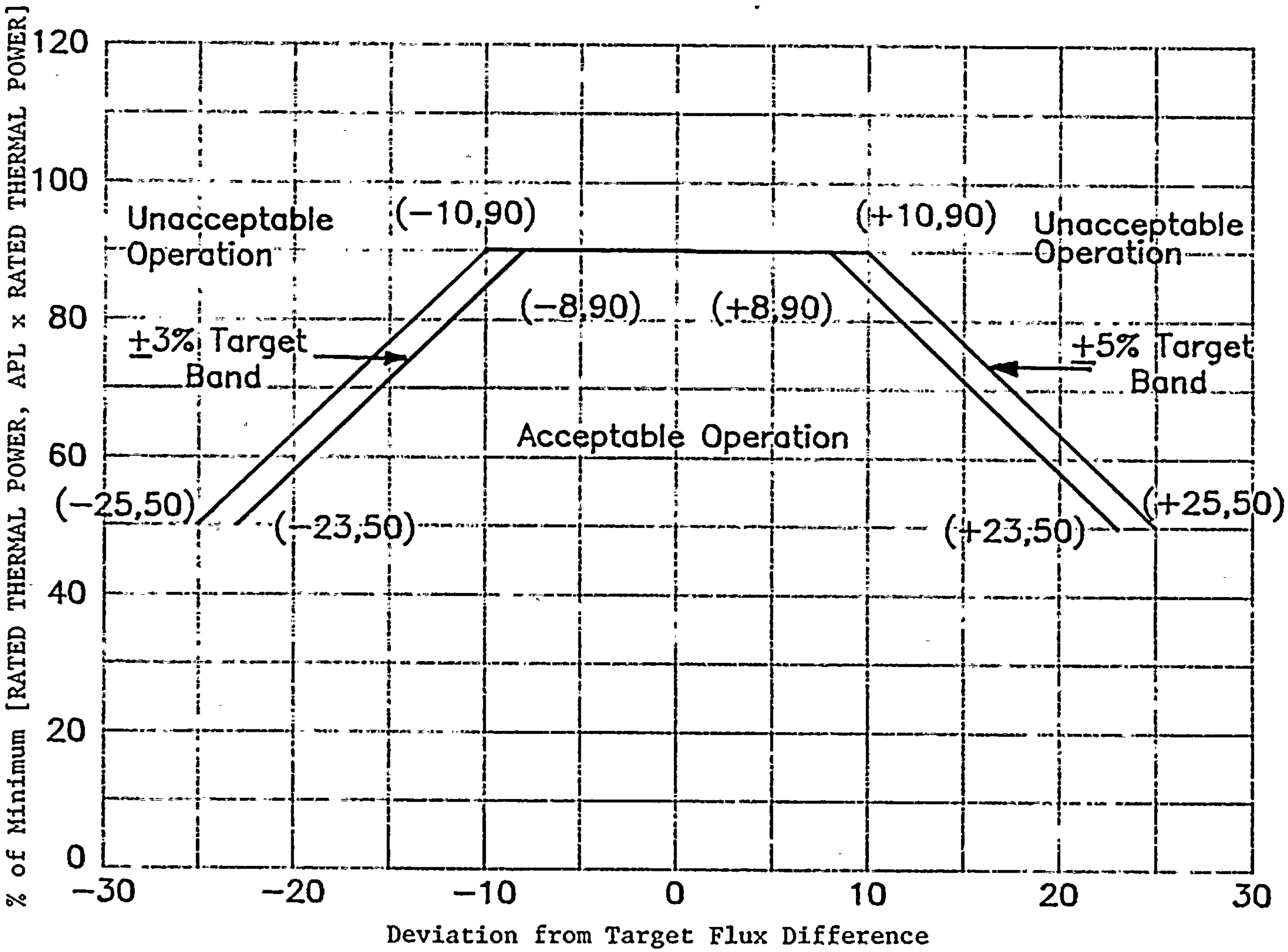

TABLE 3.2-1

DNB PARAMETERS

LIMITS

PARAMETER	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 at least three OPERABLE instrument loops.

*** 3.5% penalty for measurement uncertainty included in this value.

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, 3.4.1.2.c and 3.4.1.3.c.

* Indicated average of at least three OPERABLE instrument loops.

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:

Westinghouse Fuel

$$APL = \text{min over Z of } \frac{1.97\ K(Z)}{F_Q(Z) x V(Z) x F_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

Exxon Nuclear Co. Fuel

$$APL = \text{min over Z of } \frac{2.10\ K(Z)}{F_Q(Z) x V(Z) x F_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less.}$$

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

- V(Z) is the function defined in 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

LIMITING CONDITION FOR OPERATION (Continued)

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 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	7#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	7#
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in same loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch or 2/loop-level and 1/loop-flow mismatch	1, 2	7#

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 and below P-8	1/breaker	2	1/breaker per operating loop	1	11#
21. Reactor Trip Breakers	2	1	2	1, 2 3*,4*,5*	1, 13 14
22. Automatic Trip Logic	2	1	2	1, 2 3*,4*,5*	1 14

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow-Two Loops	S	R(8)	N.A.	1
14. Steam Generator Water Level-- Low-Low	S	R	M	1, 2
15. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	S	R	M	1, 2
16. Undervoltage-Reactor Coolant Pumps	N.A.	R	M	1
17. Underfrequency-Reactor Coolant Pumps	N.A.	R	M	1
18. Turbine Trip				
A. Low Fluid Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
B. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker				
A. Shunt Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
B. Undervoltage Trip Function	N.A.	N.A.	M(5)(11) and S/U(1)(11)	1,2,3*,4*,5*
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1,2,3*,4*,5*
23. Reactor Trip Bypass Breaker	N.A.	N.A.	M(12) and S/U(1)(13)	1,2,3*,4*,5*

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Line Pressure--Low	
a. Safety Injection (ECCS)	≤ 12.0#/24.0##
b. Reactor Trip (from SI)	≤ 2.0
c. Feedwater Isolation	≤ 8.0
d. Containment Isolation-Phase "A"	≤ 18.0#/28.0##
e. Containment Purge and Exhaust Isolation	Not Applicable
f. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
g. Essential Service Water System	≤ 14.0#/48.0##
h. Steam Line Isolation	≤ 8.0
7. Containment Pressure--High-High	
a. Containment Spray	≤ 45.0
b. Containment Isolation-Phase "B"	Not Applicable
c. Steam Line Isolation	≤ 7.0
d. Containment Air Recirculation Fan	≤ 600.0
8. Steam Generator Water Level--High-High	
a. Turbine Trip	Not Applicable
b. Feedwater Isolation	Not Applicable
9. Steam Generator Water Level--Low-Low	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
b. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0
10. 4160 volt Emergency Bus Loss of Voltage	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
11. Loss of Main Feedwater Pumps	
a. Motor Driven Auxiliary Feedwater Pumps	≤ 60.0
12. Reactor Coolant Pump Bus Undervoltage	
a. Turbine Driven Auxiliary Feedwater Pumps	≤ 60.0

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, c, and d:

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.

d. At least three of the above coolant loops shall be OPERABLE and in operation above P-12. (Refer to Technical Specification 3.3.2.1, Table 3.3-3 for instrumentation requirements.)

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 less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. With less than the number of operating coolant loops required by item d above, restore the required number of coolant loops within 2 hours or lower the reactor coolant system temperature below P-12.

LIMITING CONDITION FOR OPERATION (Continued)

d. 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.

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.

* 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.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. The 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,*

5. Residual Heat Removal - East, **

6. Residual Heat Removal - West **

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 reactor 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: 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 less than the number of operating coolant loops required by item c above, restore the required number of coolant loops within 2 hours or open the reactor trip breakers.

c. 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.

REACTOR COOLANT SYSTEM

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 $\pm$ 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 electric power circuit within one hour.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

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

**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 3.1.2.8.b.2 (MODE 4) or 3.1.2.7.b.2 (MODE 5).

REACTOR COOLANT SYSTEM

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 inoperable 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 of the inoperable PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; restore at least one of the inoperable PORVs 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.

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 929 and 971 cubic feet,

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

d. A nitrogen cover-pressure of between 599 and 644 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 12 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig.

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water,

b. Between 2400 and 2600 ppm of boron, and

c. A minimum water temperature of 80°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

   1. A day fuel tank containing a minimum volume of 70 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 42,000 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for Requirement 4.8.1.1.2.a.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.

** The provisions of Specification 4.0.6 are applicable.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% Δk/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 2400 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes** and initiate and continue boration at greater than or equal to 10 gpm of 20,000 ppm boric acid solution or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2400 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

** 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.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.* The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

* 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.

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 greater than or equal to 2000 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.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints for the OPERABLE Intermediate Range, Neutron Flux and the Power Range, Neutron Flux, Low Setpoint are set at less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than the P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating start up or PHYSICS TESTS.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference Specifications:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Unit No. 1, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Unit No. 1, Specification 3.3.3.4.

d. Fire Detection Instrumentation, Specification 3.3.3.8.

e. Fire Suppression Systems, Specifications 3.7.9.1, 3.7.9.2, 3.7.9.3 and 3.7.9.4.

f. Seismic Event Analysis, Specification 4.3.3.3.2.

g. Sealed Source leakage in excess of limits, Specification 4.7.8.1.3.

h. Moderator Temperature Coefficient, Specification 3.1.1.4

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 minor 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.

The boration capability of either system is sufficient to provide the required SHUTDOWN MARGIN from expected operating conditions after xenon decay and cooldown to 200°F. The maximum expected boration capability usable volume requirement is 3700 gallons of 20,000 ppm borated water from the boric acid storage tanks or 118,000 gallons of borated water from the refueling water storage tank. The required RWST volume is based on an assumed boron concentration of 2000 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm. The minimum contained RWST volume is based on ECCS considerations. See Section B 3/4.5.5. The boration source volume from the boric acid storage tank has conservatively been increased to 5650 gallons. This value was chosen to be consistent with Unit 1.

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.

The boron capability required below 200°F is sufficient to provide the required MODE 5 SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires usable volumes of either 4300 gallons of 20,000 ppm borated water from the boric acid storage tanks or 90,000 gallons of borated water from the refueling water storage tank. The value for the boric acid storage tank volume includes sufficient boric acid to borate to 2000 ppm. The required RWST volume is based on an assumed boron concentration of 2000 ppm. The minimum RWST boron concentration required by the post-LOCA long-term cooling analysis is 2400 ppm.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 mechnical systems and components.

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

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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.

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

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief. The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The requirement that 150 kW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensures that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.6 and 9.5 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 ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 80°F. This temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The boron concentration requirement of specification 3.9.1.b has been conservatively increased to 2400 ppm to agree with the minimum concentration of the RWST.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

Attachment 3 to AEP:NRC:0916W

SUMMARY OF DONALD C. COOK UNIT 1 AND UNIT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

SUMMARY DESCRIPTIONS FOR D. C. COOK UNIT 1 PROPOSED TECHNICAL SPECIFICATIONS

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
1-7	Definition 1.39	1	*	001	APL made a defined term.	Editorial change; definition included for clarity.
2-1	2.1.1	2	*	002	Removed reference to Figure 2.1-2 and three loop operation.	Three loop operation in Modes 1 and 2 will be prohibited.
2-3	Figure 2.1-2	2	*	003	Figure is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
2-8	Table 2.2-1	2	*	004	Parameters for three loop operation are removed.	Three loop operation in Modes 1 and 2 will be prohibited.
2-9	Table 2.2-1 Notes 3 & 4	8	*	005	Words "ΔT span" are added.	This change reflects an analysis previously submitted. See page 2 of Attachment 1 to the letter dated August 13, 1985 from M. P. Alexich to H. R. Denton (Identifier AEP:NRC:0942D). To facilitate this review, we are re-transmitting the proprietary attachment only as Attachment 4 to this letter.
3/4 1-1	3.1.1.1	1	*	006	APPLICABILITY changed to MODES 1, 2, and 3.	Editorial change to move MODE 4 SHUTDOWN MARGIN Specification to Specification 3.1.1.2.
		1		007	Mathematical symbols are written out in words.	Editorial change for clarity.
		1		007a	Specification title is changed.	Editorial change; makes the specifications of both units more similar.
3/4 1-2	4.1.1.1.1.e	1	*	008	Surveillance changed to MODE 3 only.	Editorial change to move MODE 4 SHUTDOWN MARGIN Surveillance to Surveillance 4.1.1.2.b.

NOTES: - The number in the plus sign (+) column refers to applicable section of Significant Hazards in Attachment 1.
- An asterisk in the asterisk (*) column indicates that the proposed change had been previously approved for Unit 2
- The number in the pound sign (#) column is a sequential identifier for each proposed change.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 1-3	3.1.1.2 4.1.1.2	8	*	009	Revised to include MODE 4 and MODE 5 in the same specification. Revised Technical Specification requirements based on dilution accident analysis in MODES 4 and 5.	Westinghouse Electric Corporation has performed a new analysis for D. C. Cook Unit 1 similar to that described in the letter from T. M. Anderson to V. Stello dated July 8, 1980 (Identifier NS-TMA-2273). This analysis is described in Attachment 14 to this letter. As indicated in Attachment 1 to the letter from M. P. Alexich to H. R. Denton dated March 27, 1986 (Identifier AEP:NRC:0916P), the methodology of NS-TMA-2273 has been in use on Unit 1 since beginning of Cycle 6. Attachment 1 to AEP:NRC:0916P was approved in the SER for Amendment 82 to DPR-74 To facilitate this review, we are also retransmitting Attachment 1 to AEP:NRC:0916P and NS-TMA-2273 in Attachment 14.
		1		010	Mathematical symbols are written out in words.	Editorial change for clarity.
		1		010a	Specification title is changed.	Editorial change; makes the specifications of both units more similar.
3/4 1-3a	4.1.1.2.b	1	*	011	Specification 4.1.1.2.b is moved to new page 3/4 1-3a.	Editorial change.
3/4 1-3b	Figure 3.1-3	1	*	012	New figure is added.	Editorial change.
3/4 1-3a; 3/4 1-3b		1	*	013	Pages added due to length of new specification.	Editorial change.
3/4 1-4	3.1.1.3 4.1.1.3	1		014	"reactor pressure vessel" is changed to "reactor coolant system".	Editorial change; makes the Specifications of both Units more similar.
		1		015	Mathematical symbols are written out in words.	Editorial change for clarity.

AEP:NRC:0[illegible] ATTACHMENT 3

SUMMARY DESCRIPTIONS FOR D. C. COOK UNIT 1 PROPOSED TECHNICAL SPECIFICATIONS

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		8		016	Flow rate requirement reduced to 2000 gpm.	An analysis was performed to reduce the required reactor coolant flow rate to 2000 gpm. See Attachment 5 for discussion of heat removal, mixing, and stratification considerations. See Attachment 14 for dilution transient considerations.
		4	*	017	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
3/4 1-5; 3/4 1-5a	3.1.1.4 Figure 3.1-2	8		018	The upper limit on MTC for operation above 70% RTP is changed. The upper limit is now graphically displayed (see Item 020).	To improve operational flexibility. Justification provided in Attachment 6, Item Number 4.
		1		019	Mathematical symbols are written out in words.	Editorial change for clarity.
		1		020	The new MTC limits proposed in item 018 are now graphically displayed in Figure 3.1-2.	Editorial change.
3/4 1-7	3.1.2.1	4		021	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
3/4 1-11	3.1.2.3	4	*	022	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
	ACTION c	1		022a	"ar" is changed to "are".	Editorial change; typographical error correction.
	4.1.2.3.1	1		023	Mathematical symbols are written out in words.	Editorial change for clarity.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	4.1.2.3.2.b	1		023a	Period is added.	Editorial change; typographical error correction.
3/4 1-13	3.1.2.5	4	*	024	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
	4.1.2.5.b	1		025	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 1-14	3.1.2.6	1		026	"STATUS" is changed to "status".	Editorial change; "status" is not a defined term.
	4.1.2.6.b	1		027	Mathematical symbols are written out in words.	Editorial changes for clarity.
3/4 1-15	3.1.2.7	4		028	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
				029	(No change for this identifier).	
	3.1.2.7.a.1 3.1.2.7.b.1	8	*	030	Changed BAST and RWST minimum volumes.	Boration source volumes have been adjusted to address the shutdown margin required for a dilution transient when operating on RHR at the beginning of cycle. Both volumes are usable volumes. T/S values for volumes have been selected to bound both Units. The words "borated water" are added for consistency with Unit 2. See Attachment 13. The dilution transient is discussed in Attachment 14.
	3.1.2.7.b.2	8		031	RWST minimum boron concentration is changed.	The minimum RWST boron concentration limit has been increased to provide additional margin for the LOCA long-term cooling criterion. See Attachment 13.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	3.1.2.7.b.3	11		031a	The required RWST temperature is increased to 80°F.	The minimum RWST temperature is conservatively raised to the temperature required for operability as a safeguards system in modes 1,2, 3 & 4. The value of 80°F from the Unit 2 LOCA analysis is conservatively chosen.
	4.1.2.7.b	11		031b	The RWST temperature will be monitored regardless of outside air temperature.	This is a conservative increase in surveillance requirements.
3/4 1-16	3.1.2.8.a.1	8		032	Changed BAST minimum volume.	Boration source volume has been adjusted to address the shutdown margin required for a dilution transient when operating on RHR at the beginning of cycle. The volume is a usable volume. T/S values for volumes have been selected to bound both Units. The words "borated water" are added for consistency with Unit 2. See Attachment 13. The dilution transient is discussed in Attachment 14.
	3.1.2.8.b.2	8		033	RWST minimum boron concentration is changed.	The minimum RWST boron concentration limit has been increased to provide additional margin for the LOCA long-term cooling criterion. See Attachment 13.
	3.1.2.8.b.2	8		034	RWST boron concentration upper limit is added.	The containment sump pH analysis and the changeover to hot-leg recirculation safeguards analysis require an upper limit on the RWST concentration. See Attachment 13.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	3.1.2.8.b.3	11		034a	The required RWST temperature is increased to 80°F.	The minimum RWST temperature is conservatively increased to the value for the Unit 2 LOCA analysis. The Unit 1 analysis was performed with an RWST temperature of 70°F.
3/4 1-17	4.1.2.8.b	11		034b	The RWST temperature will be monitored regardless of outside air temperature.	This is a conservative increase in surveillance requirements.
3/4 1-18; 3/4 1-19; 3/4 1-19a	3.1.3.1	1		035	The words "which are inserted in the core" are removed.	Editorial change. These words refer to part length rods inserted in the core. See p 3/4 1-14 Rev. 4, STS. There are no part length rods in Cook Unit 1.
	3.1.3.1 ACTION b	1		036	The word "bank" is replaced by the words "group step counter".	Editorial change to clarify the Specification. Unit 1 is equipped with group step counters not bank demand counters. Makes the Specifications of both units more similar.
	ACTION c	1		037	The words "due to causes other than addressed by ACTION a, above," are added.	Editorial change to clarify meaning of Specification; makes Specifications of both units more similar.
	ACTION c.1 ACTION c.2	1		038	"The rod" is changed to "The affected rod".	Editorial change for clarity.
	ACTION c.2.a	3		039	This ACTION statement is replaced.	The analyses which would require re-evaluation if Unit 1 were to be operated with an inoperable control rod are more numerous than those requiring re-evaluation in the current specifications. The change makes the Unit 1 specifications more like the Unit 2 specifications. Also see STS Rev. 4, pp 3/4 1-14 and 3/4 1-16.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	ACTION c.2.c	3		040	This ACTION statement is added.	Additional power distribution monitoring would be required if Unit 1 were to be operated with an inoperable control rod. The change makes the Unit 1 specifications more like the Unit 2 specifications. Also see STS Rev. 4, p 3/4 1-15.
	ACTION c.2.d ACTION c.2.e	1		041	Current ACTION statements c.2.c and c.2.d are renumbered.	Editorial change made to reflect addition of new ACTION c.2.c.
		1		042	Words added to emphasize that when ACTION c.2 is chosen that items a, b and c must be performed plus the choice of either d or e.	Editorial change. These clarifications also makes the Specifications of both units more similar.
	ACTION c.2.d	1		043	Mathematical symbols are written out in words.	Editorial change for clarity.
	ACTION c.2.e	2		044	Reference to Figure 3.1-2 is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
	Table 3.1-1	1		045	Table referred to in Item 039 is added.	Editorial change. See Item 039.
3/4 1-21	3.1.3.3	1		046	Mathematical symbols are written out in words.	Editorial change.
		1		047	"(228 steps)" is added.	Editorial change; clarifies meaning of fully withdrawn.
		1		048	APPLICABILITY changed to MODES 1 and 2.	Editorial change; The current Technical Specification incorrectly indicate the applicable MODE to be 3. The specification is applicable to plant operation in MODES 1 or 2. The surveillance test is performed in MODE 3. Makes the Unit 1 Specifications more like the Unit 2 Specifications.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		2		049	ACTION statement b removed.	Three loop operation in Modes 1 and 2 will be prohibited.
	4.1.3.3	3		050	words "prior to entering MODE 2" replace "prior to reactor criticality".	Requiring the completion of this test prior to entering MODE 2 is conservative to requiring the test prior to criticality. MODE 2 is entered with the reactor subcritical by 1%. However, making the requirement mode dependent eases administrative control.
3/4 1-22	3.1.3.4	1		051	"(228 steps)" is added.	Editorial change, clarifies meaning of fully withdrawn.
		1		052	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 1-23	3.1.3.5	2	*	053	Reference to Figure 3.1-2 is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
	ACTION b	1		054	"figures" becomes "figure".	Editorial change.
		1		055	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 1-24; 3/4 1-25; 3/4 1-26	Figure 3.1-1 Figure 3.1-2	1	*	056	Rod Group Insertion Limits figure for 4 Loop Operation is redrawn with labeled endpoints.	Editorial change. See memo dated February 26, 1986, from F. J. Silva to J. C. Miller of Westinghouse Electric Corporation found in Attachment 7.
		2	*	057	Rod Group Insertion Limit figure for 3 Loop Operation is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
		1		058	Rod Group Insertion Limit figure for 4 Loop Operation is renamed Figure 3.1-1.	Editorial change.
		1		059	Pages 3/4 1-25 and 3/4 1-26 are removed.	Editorial change; blank pages are unnecessary at the end of a section.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 2-1	3.2.1	1		060	"3.4.2" becomes "3/4.2" in title.	Editorial change.
		1		061	APL footnote is removed.	Editorial change. APL is now found in definitions.
		1		062	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 2-2	3.2.1.a.2.c	10	*	063	Exemption from AFD requirements for APDMS calibration is removed.	The Axial Power Distribution Monitoring System (APDMS) is not used. The plant will operate below the Allowable Power Level (APL).
	3.2.1.d	1		064	Action d is removed.	Editorial change. This action referred to the LER section of Technical Specifications. The LER rules are now included in CFR.
3/4 2-3	4.2.1.3 4.2.1.4	10	*	065	$F_Q^M(Z)$ is changed to APL. Referenced specification number has changed.	The combined $F_Q(Z)$ - target flux surveillance changed to combined APL - target flux surveillance. See Technical Specification 3.2.6.
3/4 2-4	Figure 3.2-1	1		066	Figure is redrawn.	Editorial change for clarity.
3/4 2-5	3.2.2	1	*	067	Description of $F_Q(Z)$ penalties moved from surveillance to LCO.	Editorial change for clarity.
3/4 2-5	3.2.2	10		068	The F_Q limit for Exxon fuel is changed to fixed value of 2.04.	This change is based on the Exxon analysis presented in XN-NF-85-115(P), Rev. 2. This report was transmitted to the NRC with a letter dated January 15, 1987 from Exxon Nuclear Company, Inc. The Exxon letter was identified as GNW:001:87. This report was placed on our docket by a letter dated January 29, 1987 from M. P. Alexich to the NRC Document Control Desk. (Identifier AEP:NRC:0940E). The new analysis does not result in a

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						burnup dependence for Exxon fuel as discussed in Section 2.0 of XN-NF-85-115(P). This result is also discussed and supported in a letter from H. G. Shaw of Advanced Nuclear Fuels to Mr. Rick Bennett dated March 5, 1987, identifier ENC/AEP 0556. The letter from Mr. Shaw is included as Attachment 15. To facilitate this review we are retransmitting AEP:NRC:0940E and a proprietary version only of XN-NF-85-115(P) with Attachment 15. In addition, we are retransmitting our letter AEP:NRC:1018 and its Attachment 1 and its proprietary Attachment 4 with Attachment 15 of this letter. These documents demonstrate our recognition of burnup limits based on mechanical design and our commitment not to exceed those limits without performing required analyses.
		1	*	069	Definitions for P, $F_Q(Z)$, and K(Z) reworded with no change of meaning.	Editorial change for clarity.
	3.2.2.a	10	*	070	Modified existing ACTION statement a.1 to remove the requirement to lower the Overpower ΔT (OPΔT) in hot standby.	In the current Technical Specification 3.2.2, ACTION a.1 requires that the OPΔT trip setpoint reduction be performed when the reactor is in hot standby. This has been deleted. The change in the ACTION statement for specification 3.2.2 is consistent with the draft version of the Westinghouse Standardized Technical Specifications, Revision 5. Our evaluation indicated that the reduction of the Overpower ΔT setpoint can be done while the reactor is in Mode 1.
		1		071	"F_Q" is changed to "$F_Q(Z)$".	Editorial change for clarity.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		10	*	072	ACTION 3.2.2.a.2 is removed.	The APDMS is not used. The plant will operate below APL.
3/4 2-5; 3/4 2-6	3.2.2	1		073	ACTION statement b is moved from page 3/4 2-6 to page 3/4 2-5.	Editorial change for clarity.
3/4 2-6; 3/4 2-7; 3/4 2-8; 3/4 2-9	4.2.2.2	10	*	074	Much of this surveillance requirement has been moved to APL Specification 3.2.6.	Specification is simplified. The requirements that were in this specification are now incorporated in specification 3.2.6. No provisions of current Technical Specifications other than those pertaining to the following were deleted or substantially modified: (1) APDMS - See items 63, 72, and 134. (2) Exxon F_Q limit, based on the revised LOCA analysis - See item 68. (3) Removal of burnup dependencies for F_Q. The justification for removing the implied burnup limit of 42.2 MWD/Kg for Westinghouse fuel is contained in the group 10 of the significant hazards evaluation, Attachment 1 of This submission. See items 68 (Exxon fuel) and group 10 of Attachment 1 (Westinghouse fuel). (4) Removal of the V(Z) figure - See item 75. (5) The modification to existing ACTION statement a.1 of Technical Specification 3.2.2 - See items 70 and 96. (6) Items 65, 74, and 96 describe the simplification. Item 189 adds the requirement to submit the Peaking Factor Limit Report.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						The methodology which supports the F_Q surveillance is described in Part B of the Westinghouse topical report, WCAP-10217-A "F_Q Surveillance Technical Specification". Attachment 19 includes: (1) A review of proposed simplifications by our fuel vendor, Westinghouse. (2) A letter from our fuel vendor, Westinghouse, supporting a burnup Independent F_Q for Westinghouse fuel to at least 60 MWD/Kg peak pellet burnup.
3/4 2-8(a)	Figure 3.2-3	1		075	The V(Z) function provided by Exxon Nuclear Co. is removed from Technical Specifications. This page is to be removed from T/S.	Editorial changes; The V(Z) curve in Figure 3.2-3 is associated with the previous fuel vendor's methodology. The equivalent penalty is supplied by the current fuel vendor, Westinghouse, in the Peaking Factor Limit Report. Removal of this figure was previously proposed for Cycle 8 operation. Removal of this page was inadvertently omitted from Amendment 74 to DPR-58.
3/4 2-10	Figure 3.2-2	1	*	076	The figure is redrawn.	Editorial change for clarity.
		1		077	The page number is changed to 3/4 2-7.	Editorial change.
3/4 2-11	Figure 3.2-3	1		078	The figure is redrawn.	Editorial change for clarity.
		1		079	The page number is changed to 3/4 2-8.	Editorial change.
3/4 2-12	3.2.3	1		080	Mathematical symbols are written out in words.	Editorial change for clarity.
		1		081	"power" is changed to "POWER".	Editorial change. Thermal power is a defined term.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		1		082	The page number is changed to 3/4 2-9.	Editorial change.
3/4 2-13	4.2.3	1		083	4.2.3.1 is changed to 4.2.3.	Editorial change.
		1		084	The page number is changed to 3/4 2-10.	Editorial change.
3/4 2-14; 3/4 2-15	3/4.2.4	1		085	The page numbers are changed to 3/4 2-11 and 12, respectively.	Editorial change.
		1		086	"LIMITS" is added to title.	Editorial change.
		1		087	Mathematical symbols are written out in words.	Editorial change.
	ACTION b.2	1		087a	"trip" is changed to "Trip".	Editorial change.
3/4 2-16	3/4.2.5	1		088	The page number is changed to 3/4 2-13.	Editorial change.
	4.2.5.2 4.2.5.3	3	*	089	Surveillance Requirement 4.2.5.2 is expanded and clarified.	Surveillance requirements revised to add CHANNEL CALIBRATION and flow measurement once per 18 months. The 18-month calibration and flow measurement are required to ensure the accuracy of the 12-hour surveillance of RCS flow and the accuracy of the low flow trips. Monthly flow surveillance is removed as redundant to shiftly surveillance. Resulting surveillance requirements are consistent with Unit 2 Technical Specifications. See Attachment 6, Item Number 10.
	4.2.5.4	7	*	090	Exemption from Specification 4.0.4 is added for primary flow surveillances.	Primary flow surveillances must be made in the applicable mode.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 2-17	Table 3.2-1	2	*	091	The parameters for three loop operation are removed.	Three-loop operation in Modes 1 and 2 will be prohibited.
		1		092	The parameters for Design Thermal Power are removed.	Editorial change for simplification; these values cannot be used prior to completion of power re-rating analysis.
		1		093	Units used for pressure changed from psia to psig.	Editorial change for simplification.
		1		093a	Exponent changed from 10^8 to 10^6.	Editorial change. 1.386×10^8 changed to 138.6×10^6 for consistency with the Unit 2 T/Ss.
		8		094	Footnotes are added for RCS T_{avg} and RCS Total Flow Rate.	This change reflects an analysis previously submitted. See page 3 of Attachment 1 to the letter dated August 13, 1985 from M. P. Alexich to H. R. Denton (Identifier AEP:NRC:0942D). To facilitate this review, we are re-transmitting the proprietary attachment only as Attachment 4 to this letter. See also Attachment 6, Item Number 9, for supplementary information supplied by our contractor Westinghouse. The details of the calculation for Unit 2 are exhibited on page (vii) of Attachment 18.
		1		095	This page number is changed to 3/4 2-14.	Editorial change.
3/4 2-18; 3/4 2-19; 3/4 2-20; 3/4 2-21; 3/4 2-22; 3/4 2-23; 3/4 2-24	3.2.6	10	*	096	This entire Technical Specification is changed to an Allowable Power Level (APL) Technical Specification.	The APDMS, an option in current Technical Specifications, is not used. The plant will operate below APL. This specification is added to satisfy the requirements of the Westinghouse F_Q Surveillance Technical Specification Methodology. No provisions of current Technical Specifications other than those

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS

pertaining to the following were deleted or substantially modified:

(1) APDMS - See items 63, 72, and 134.

(2) Exxon F_Q limit, based on the revised Exxon LOCA analysis - See item 68.

(3) Removal of burnup dependencies for F_Q. The justification for removing the implied burnup limit of 42.2 MWD/Kg for Westinghouse fuel is contained in group 10 of the significant hazards evaluation, Attachment 1 of this Submission. See items 68 (Exxon fuel) and group 10 Attachment 1 (Westinghouse fuel).

(4) Removal of the V(Z) figure - See item 75.

(5) The modification to existing ACTION statement a.1 of Technical Specification 3.2.2 - See items 70 and 96.

(6) Items 65, 74, and 96 describe the simplification. Item 189 adds the requirement to submit the Peaking Factor Limit Report.

The methodology which supports the F_Q surveillance is described in Part B of the Westinghouse topical report, WCAP-10217-A "F_Q Surveillance Technical Specification". Attachment 19 includes:

(1) A review of proposed simplifications by our fuel vendor, Westinghouse.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						(2) A letter from our fuel vendor, Westinghouse, supporting a burnup Independent F_Q for Westinghouse fuel to at least 60 MWD/Kg peak pellet burnup. Requirements which were formerly in specification 3.2.2 have been incorporated in this specification. In the current Technical Specification 3.2.2, ACTION a.1 requires that the OPΔT trip setpoint reduction be performed when the reactor is in hot standby. This has been deleted. The change in the action statement for specification 3.2.2 is consistent with the draft version of the Westinghouse Standardized Technical Specifications, Revision 5. Our evaluation indicated that the reduction of the Overpower ΔT setpoint can be done while the reactor is in Mode 1.
		1		097	The new APL Technical Specification is on pages 3/4 2-15 and 3/4 2-16.	Editorial change.
3/4 2-17 through 3/4 2-24		1		098	Pages 3/4 2-17 through 3/4 2-24 are deleted.	Editorial change.
3/4 3-2	Table 3.3-1	1		099	This page is intentionally left blank.	Editorial change; Table 3.3-1 is condensed. Resulting table is more similar to the Unit 2 Technical Specifications.
3/4 3-3	Table 3.3-1 Item 2	3	*	100	Power Range, Neutron Flux Functional Unit has an added applicable mode:*.	The Plant Transient Analysis requires the Power Range, Neutron Flux Functional Unit to be operable with the reactor trip breakers in the closed position and the control rod drive

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						mechanism capable of rod withdrawal. This is consistent with section 14.3.1 of Appendix 14.C of the Unit 1 FSAR, as well as Table 4.3-1 of the Unit 1 Technical Specifications.
	Item 3	1		101	Comma is added.	Editorial change.
	Item 5	1		102	"Intermediate Range, Neutron Flux" is typed onto two lines.	Editorial change.
	Item 7	2	*	103	References to three loop operation are removed.	Three loop operation in Modes 1 and 2 will be prohibited.
	Item 8	2	*	104	References to three loop operation are removed.	Three loop operation in Modes 1 and 2 will be prohibited.
3/4 3-4	Item 13	1		105	"in" added.	Editorial change for clarity.
	Item 14	1		106	"loops" is changed to "loop".	Editorial change; grammatical error correction.
3/4 3-5	Item 16	1		107	slash replaced by hyphen.	Editorial change; typographical error correction.
	Item 20B	7	*	108	Reactor Coolant Pump Breaker Position Trip Above P-7 has an added exemption from 3.0.4 applicability.	The Reactor Coolant Pump Breaker Position Trip provides protection against DNB at reactor coolant flow rates above the P-7 interlock. This interlock is enabled between 0 and 11% rated thermal power. Technical Specification 3.4.1.1 requires all reactor coolant loops be in operation for MODES 1 and 2. With all coolant loops in operation, there is more than enough flow for DNB protection up to the P-7 interlock (11% RTP) and the ESF actuation for DNB protection is not needed in MODE 1 until after the P-7 is enabled. At that point, the Reactor

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						Coolant Pump Breaker Position Trip channel must be in operation. The proposed change to exempt Section 3.0.4 will allow entry into Mode 1 without these channels required operable but will not allow operation above P-7 interlocks without meeting the appropriate action statements. This proposed change was also recognized in later revisions to the Standard Technical Specifications.
	Item 22	1		109	Clarifications made to properly identify which ACTION statements apply to applicable mode.	Editorial change; this change corrects a format error made in the issuance of Amendment No. 99.
3/4 3-6	Table 3.3-1 Notation	2	*	110	Footnote ** is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
	ACTION 2.b	1		111	Words "of the other channels" are added.	Editorial change for clarification. Makes Specifications for both units more similar.
	ACTION 2.c	1		112	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 3-8	Table 3.3-1 ACTION 14	1	*	113	"OPEARABLE" is changed to "OPERABLE".	Editorial change; typographical error correction.
	ACTION 9	2	*	114	Action 9 is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
		1		115	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 3-9	Table 3.3-1	1		116	Mathematical symbols are written out in words.	Editorial change for clarity.
		11		117	The value of P-8 is changed to 31% RTP.	To achieve greater consistency with Unit 2 Technical Specifications. 31% is conservative relative to current 51%.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 3-12	Table 4.3-1 Item 2	3	*	118	Power Range, Neutron Flux Functional Unit has an additional Channel Functional Test (S/U(1)).	The Plant Transient Analysis requires the Power Range, Neutron Flux Functional Unit to be operable with the reactor trip system breakers in the closed position and the control rod drive mechanism capable of rod withdrawal. See section 14.3.1 of Appendix 14.C of the Unit 1 FSAR.
3/4 3-12; 3/4 3-13	Item 2, 5, 6, 7, 8, 12 & 13	7	*	119	Power, Intermediate, and Source Range Neutron Flux, Loss of Flow Single Loop and Two Loop Functional Units have added exemptions from Specification 4.0.4. Overpower ΔT and Overtemperature ΔT Functional Units have added exemptions from Specification 4.0.4 for $f_1(\Delta I)$ and $f_2(\Delta I)$ penalties.	Exemptions are provided for surveillances which must be performed in the applicable mode. Note that the * does not apply to loss of flow in two units which was inadvertently omitted from our Unit 2 submittal.
3/4 3-14	Table 4.3-1 Notation	1		120	Mathematical symbols are written out in words.	Editorial change for clarity.
		7	*	121	Footnotes (8) and (9) are added.	See remarks for Item 119.
3/4 3-16; 3/4 3-17; 3/4 3-18; 3/4 3-20; 3/4 3-21	Table 3.3-3 Item 1.e 1.f 4.d	2	*	122	References to three loop operation in Modes 1 and 2 are removed.	Three loop operation in Modes 1 and 2 will be prohibited.
		5	*	123	Reference to ### footnote for Differential Pressure Between Steam Lines-High Functional Unit changed to #### footnote.	The Differential Pressure Between Steam Lines-High actuation differs from other ESF Actuation signals in that a signal from one loop is compared to signals in the other loops. Placing all channels associated with the idle loop in trip would result in an ESF actuation. This actuation would preclude 3 loop operation. Therefore, the appropriate channels to trip are the bistables which indicate

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						low active steam pressure relative to the idle loop. This action reduces the ESF actuation logic for the active loop differential pressures from 2/3 to 1/2. An ESF actuation does not result because the three bistables, which indicate low idle loop steam pressure relative to the active loops, and which are in a 2/3 logic, are not tripped. See simplified logic diagram in Attachment 16.
	Items 1.f & 4.d	12		124	References to Footnote ** are removed.	This change reflects an analysis previously submitted. See Attachment 4 of the letter dated August 13, 1985 from M. P. Alexich to H. R. Denton (Identifier: AEP:NRC:0942D). To facilitate this review we are retransmitting the proprietary attachment only as Attachment 8 to this letter.
				125	(No change for this identifier)	
3/4 3-22	Table 3.3-3	5	*	126	Footnote #### is added.	See remarks for Item 123.
		12		127	Footnote ** is removed.	See remarks for Item 124.
3/4 3-23	Table 3.3-3	9	*	128	Reworded Condition and Setpoint, Function description for P-12 interlock.	This change clarifies the definitions of the interlock and makes the definition less ambiguous. Patterned after STS, Rev.4.
		1		129	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 3-29	Table 3.3-5 Item 8a	12		130	Reactor trip is removed from description.	This wording is consistent with STS, Revision 4. The analysis of Excessive Heat Removal due to Feedwater System Malfunctions event is the only analysis which uses the ESF Steam

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						Generator Water Level-High High feature. This analysis uses the reactor trip on turbine trip as an anticipatory trip to terminate the event. Since the trip is not required the only response time needed is the response time for turbine trip. This event is discussed in greater detail in item 2 of Attachment II to Attachment 6 of this submittal.
3/4 3-31	Table 4.3-2 Item 1c	1		131	Period is changed to comma.	Editorial change; typographical error correction.
3/4 3-33	Table 4.3-2 Item 6d	1	*	132	Loss of Main Feedwater Pumps Mode 3 Surveillance Requirement deleted.	Editorial change; Mode 3 applicability for Loss of Main Feedwater Pumps was deleted from Table 3.3-3 in Unit 1 License Amendment #92.
3/4 3-33a	Table 4.3-2 Item 8.b	1		133	"Loss of Voltage" is changed to "Degraded Voltage".	Editorial change to clarify difference between Item 8a & 8b.
3/4 3-49; 3/4 3-50	3.3.3.6 4.3.3.6	10	*	134	This entire Technical Specification is removed.	The APDMS is not used. The plant will operate below APL.
3/4 4-2	3.4.1.2	3	*	135	Criterion for the operability of reactor coolant loops are established based on the status of the reactor trip system breakers and/or the control rod system.	The Plant Transient Analysis requires these changes based on the uncontrolled control rod bank withdrawal from subcritical. The proposed Specification conservatively requires 3 pumps for consistency with Unit 2. An appropriate ACTION statement has been proposed to correspond to this requirement. See letter from E. P. Rahe, Jr. to D. Eisenhut dated July 9, 1984 (Identifier NS-TA-84-003) and letter from M. P. Alexich to H. R. Denton dated July 30, 1984 (Identifier AEP:NRC:0895). To facilitate this review, we are retransmitting these letters as Attachment 17.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		11		135a	Criterion for the operability of reactor coolant loops based on P-12 is added.	Table 3.3-3 requires at least three loops operating above P-12. This ensures flow through RTD by-pass loops. This provision is added for consistency with Table 3.3-3. An appropriate ACTION statement has been proposed to correspond to this requirement.
		1		136	Existing text reorganized for convenience. ACTION b becomes ACTION d.	Editorial change.
	3.4.1.2 ACTION d Footnote *	4	*	137	** is added to ACTION d and footnote *; footnote ** is added to bottom of page.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
3/4 4-2a	4.4.1.2.1 4.4.1.2.2	1	*	138	Surveillances, footnotes and ACTION d moved from previous page.	Editorial change; additional text requires moving this material.
3/4 4-3; 3/4 4-3a	3.4.1.3	3	*	139	Criteria for the operability of reactor coolant loops are established based on the status of the reactor trip system breakers and/or the control rod system.	See remarks for Item 135.
		1		140	Existing text reorganized for convenience. ACTION b becomes ACTION c.	Editorial change.
	3.4.1.3 ACTION c Footnote ***	4	*	141	**** is added to ACTION c and footnote ***; footnote **** is added on page 3/4 4-3a.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
		1	*	142	Footnotes are moved from page 3/4 4-3 to page 3/4 4-3a.	Editorial change; expanded specification requires the movement of this material.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		1		142a	Changed 62.00% to 62%. Removed underlining.	Editorial change.
3/4 4-3b; 3/4 4-3c; 3/4 4-3d	3.4.1.4 4.4.1.4	2	*	143	The entire Technical Specification is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
		1		144	Pages 3/4 3-c and 3/4 3-d are to be removed.	Editorial change.
3/4 4-4	3.4.2	4		145	Footnote ** added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
		11		146	Footnote * is added.	This change clarifies the conditions to which the pressurizer code safety valve lift settings correspond. This footnote is in the Unit 2 Technical Specifications and in practice accurately describes what is done currently in Unit 1. This change does not impact the operations of Unit 1 and is primarily administrative in nature.
	ACTION b	3	*	147	ACTION statement added.	Changed to make the Specifications of both Units more similar. The analyses of overpressurization for Unit 2 described in XN-NF-85-28(P), Supplement 1 "D. C. Cook Unit 2, Cycle 6 Safety Analysis Report", identified the need for the proposed additional ACTION to prevent overpressurization with no safety valve operable. Since Unit 1 and Unit 2 primary systems are essentially identical, the additional ACTION is proposed for Unit 1.
3/4 4-5	3.4.3 4.4.3	11		148	Footnote * is added.	See remarks for Item 146.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 4-35	3.4.11	6	*	149	ACTION changed to only allow one PORV or block valve inoperable. Making more than one PORV inoperable without shutting down the reactor is not allowed.	Changed to make the Specification of both units more similar. The proposed changes are intended to ensure that the PORVs are available to assist in RCS depressurization following a steam generator tube rupture without offsite power. See Section 14.2.4, "Steam Generator Tube Rupture", of the Unit 1 FSAR.
		1	*	150	Reference to Section 6.9.1.9 is deleted.	This reference is no longer appropriate. Section 6.9.1.9 of the Technical Specifications delineated reportability requirements prior to including these requirements in 10 CFR 50.72 and 10 CFR 50.73.
3/4 4-36	4.4.11.1	1	*	151	Portions of expanded ACTION statement and surveillance requirements moved to p 3/4 4-36.	Editorial change; surveillance requirement moved from previous page to this page.
	4.4.11.2	1	*	152	Reference to Section 6.9.1.9 is deleted.	See remarks for Item 150.
		1		153	Footnote * is changed to **.	Editorial change.
	4.4.11.3	1	*	154	Reference to Surveillance 4.8.2.3.2.c is changed to 4.8.2.3.2.d.	Editorial change; the current reference is incorrect.
3/4 5-1	3.5.1.b	1		155	Text revised.	Editorial change to make the specifications of both units more similar.
	3.5.1.c	8		156	Minimum accumulator boron concentration is changed.	The minimum accumulator boron concentration limit has been increased to provide additional margin for the LOCA longterm cooling criterion. See Attachment 13.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	3.5.1.c	8		157	Accumulator boron concentration upper limit is added.	The containment sump pH analysis and the changeover to hot-leg recirculation safeguards analysis require an upper limit on the accumulator concentration. See Attachment 13.
3/4 5-11	3.5.5.b 4.5.5.a.1	1		158	Text revised.	Editorial change to make the specifications of both units more similar.
				159	(No change for this identifier).	
	3.5.5.b	8		160	Minimum RWST boron concentration is changed.	The minimum RWST boron concentration limit has been increased to provide additional margin for the LOCA long-term cooling criterion. See Attachment 13.
		8		161	RWST boron concentration upper limit is added.	The containment sump pH analysis and the changeover to hot-leg recirculation safeguards analysis require an upper limit on the RWST concentration. See Attachment 13.
	3.5.5.c	11		161a	The required RWST temperature is increased to 80°F.	The minimum RWST temperature is conservatively increased to the value for the Unit 2 LOCA analysis. The Unit 1 analysis was performed with an RWST temperature at 70°F.
	4.5.5.b	11		161b	The RWST temperature will be monitored regardless of outside air temperature.	This is a conservative increase in surveillance requirements.
3/4 7-1	3.7.1.1	2	*	162	ACTION b is modified to remove three loop operation in Modes 1 and 2.	Three loop operation in Modes 1 and 2 will be prohibited.
3/4 7-3	Table 3.7-2	2	*	163	Table is removed.	Three loop operation in Modes 1 and 2 will be prohibited.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 7-4	Table 4.7-1	11		163a	Footnote * is added.	This change clarifies the conditions to which the pressurizer code safety valve lift settings correspond. This footnote is in the Unit 2 Technical Specifications and in practice accurately describes what is done currently in Unit 1. This change does not impact the operations of Unit 1 and is primarily administrative in nature.
3/4 7-5	4.7.1.2	8	*	164	Discharge pressures for auxiliary feedwater pump flow testing changed.	The limiting accident for auxiliary feedwater pump performance is the feedwater line break. In Amendment 82 to DPR 74 (Unit 2), the auxiliary feedwater pump discharge pressures were lowered to the values being proposed for Unit 1. This reduction was based on the feedwater line break analysis performed by Exxon Nuclear Co., which is found in Section 15.2.8 of XN-NF-85-64 (P), Rev. 1, "Plant Transient Analysis for D. C. Cook Unit 2 with 10% Steam Generator Tube Plugging". This new analysis allowed credit for operator action after 10 minutes to isolate the faulted steam generator and ensure adequate auxiliary feedwater was delivered to the intact steam generators. This differed from the previous Unit 2 analysis, which assumed auxiliary feedwater was delivered within one minute following the initiation of the break. The new Exxon analysis resulted in reduced auxiliary feedwater discharge pressure requirements, which were reflected in the Amendment 82 T/Ss.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						For Unit 1, Feedwater Line Break is not part of the license basis, as noted in Chapter 14.2.8.1 of the Unit 1 UFSAR. However, an evaluation of this accident was performed and included in Chapter 14.2.8.1 of the UFSAR. This analysis, like the Exxon analysis, assumed 10 minutes for operator action and an identical value for the amount of auxiliary feedwater delivered to the intact steam generators (600 gpm). Thus, it supports the same value for auxiliary feedwater pump discharge pressure as that currently included in the Unit 2 T/Ss, and the change is requested to maintain consistency between the Units.
		1		165	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 7-10	3.7.1.5	3	*	166	ACTION statements are revised.	The provision of the ACTION statement for MODE 1 permitting operation in MODE 1 with a steam generator stop valve closed is deleted. Failure to restore the stop valve to operable status in MODE 1 results in MODE 2 instead of MODE 4 operation. The reference in the MODE 2, 3 ACTION statement to continued operation in MODE 1 is deleted. The STS terminology is changed to be consistent with Cook Plant terminology.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						The proposed Technical Specification achieves substantially greater consistency with STS, Rev.4. The exemption from 3.0.4 permits entry into MODES 2 and 3 with an inoperable stop valve because such operation is permitted in those modes.
	4.7.1.5.1	1	*	167	Specification 4.7.1.5 is renumbered 4.7.1.5.1.	Editorial change.
	4.7.1.5.2	7	*	168	Exemption from Specification 4.0.4 is added for entry into Mode 3.	Exemptions are provided for surveillances which must be performed in the applicable mode.
		7	*	169	Exemption from Specification 4.0.4 is provided for entry into Mode 2 with stop valves closed for PHYSICS TESTS.	This specification ensures that no more than one steam generator will blowdown in the event of steam line rupture. If the valves are closed during PHYSICS TESTS only the affected steam generator can blowdown. This provision provides added operational flexibility at BOC.
3/4 8-5	3.8.1.2	4		170	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
		1		171	Footnote * is changed to **.	Editorial change.
3/4 9-1	3.9.1	1		172	Mathematical symbols are written out in words.	Editorial change for clarity.
		4		173	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	3.9.1.b ACTION	11		173a	The required boron concentration for refueling is increased to 2400 ppm.	The required concentration is conservatively increased to agree with the RWST concentration. The result is a substantial increase in the amount by which the core is shutdown during refueling.
3/4 9-2	3.9.2	4		174	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
3/4 9-9	3.9.8.1 4.9.8.1	1		175	Mathematical symbols are written out in words.	Editorial change for clarity.
		8		176	Flow rate requirement reduced to 2000 gpm.	An analysis was performed to reduce the required reactor coolant flow rate to 2000 gpm. See Attachment 5 for discussion of heat removal, mixing, and stratisfications considerations. See Attachment 14 for dilution transient considerations.
		4	*	177	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
3/4 10-2	4.10.2.2	1	*	178	Referenced specifications are renumbered.	Editorial change; reflects simplification of F_Q and APL specifications, 3.2.2 and 3.2.6 respectively. See page 3/4 2-6.
		1		179	Mathematical symbols are written out in words.	Editorial change for clarity.
		1		180	Reference to the Augmented Startup Test Program is removed.	Editorial change; the Augmented Startup Test Program has been completed. See Attachment 6, Item Number 11.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 10-3	3.10.3.b	12		181	Specification is reworded.	Clarifies intention of specification. See Attachment 6, Item Number 14.
		1		182	"reactor trip setpoints" is re-written as "Reactor Trip Set-points".	Editorial change for consistency with specification 3.10.5.b.
	3.10.3.b ACTION 4.10.3.2	1		183	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 10-5	3.10.4.b	12		184	Specification is reworded.	See Remarks for Item 181.
		1		185	"reactor trip setpoints" is re-written as "Reactor Trip Set-points.	Editorial change for consistency with specification 3.10.5.b.
	ACTION	1		185a	"THERMAl" is changed to "THERMAL".	Editorial change; typographical error correction.
	3.10.4.b ACTION 4.10.4.1	1		186	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 10-6	3.10.5.b	12		187	Specification is reworded.	See Remarks for Item 181.
		1		188	Mathematical symbols are written out in words.	Editorial change for clarity.
	4.10.5.1	1		188a	"the" is added.	Editorial change.
6-19	6.9.1.11	10		189	Section added.	The Peaking Factor Limit Report will be submitted each cycle. This achieves greater consistency with STS, Rev. 4. See Attachment 9 for reason for change from 60 days to 15 days. This item is specifically addressed in both the cover letter of this submission and our significant hazards evaluation in Attachment 3.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
B 2-1; B 2-1a	2.1.1 (Bases)	B	*	190	References to three loop operation and Figure 2.1-2 are removed.	Three loop operation in Modes 1 and 2 will be prohibited.
		B		191	Headings are clarified; footnotes are added.	Editorial change to clarify meaning of text.
B 2-5	Overtemperature ΔT (Bases)	B	*	192	Paragraph referring to three loop operation is removed.	Three loop operation in Modes 1 and 2 will be prohibited.
	Overpower ΔT (Bases)	B	*	193	Added reference to $f(\Delta I)$ penalty for $OP\Delta T$.	Penalty is used for the current analysis. Included in the basis for completeness and consistency with Unit 2 Technical Specification Bases.
	Pressurizer Pressure (Bases)	B	*	194	Added reference to the use of the pressurizer pressure high trip in the loss of load event.	See section 14.C.3.6 of Unit 1 FSAR.
B 2-5; B 2-6	2.2.1 (Bases)	B	*	195	Moved text from page B 2-6 to B 2-5.	Editorial change.
B 2-6	Loss of Flow (Bases)	B	*	196	The value of the P-8 setpoint is changed to 31%. This sentence is reworded.	Editorial change.
		B	*	197	References to three loop operation are removed.	Three loop operation in Modes 1 and 2 will be prohibited. See Attachment 6, Item Number 1.
		B		198	Mathematical symbols are written out in words.	Editorial change for clarity.
B 2-6; B 2-7; B 2-8	2.2.1 (Bases)	B	*	199	Moved text from page B 2-7 to B 2-6, and from page 2-8 to B 2-7. Page B 2-8 may now be deleted.	Editorial change.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
B 3/4 1-1	3/4.1.1.1 3/4.1.1.2 (Bases)	B	*	200	Revision to Shutdown Margin Basis.	Bases revised to address dilution transient when operating on RHR at beginning of cycle. See Attachment 14.
		B		200a	350°F is changed to 200°F.	Editorial change; typographical error correction. The upper limit to Mode 5 is 200°F.
		B		201	Mathematical symbols are written out in words.	Editorial change for clarity.
	3/4.1.1.3 (Bases)	B		202	Flow rate requirement reduced to 2000 gpm.	An analysis was performed to reduce the required reactor coolant flow rate to 2000 gpm. See Attachment 5 for discussion of heat removal, mixing, and stratification considerations. See Attachment 14 for dilution transient considerations.
		B		202a	Circulation time is increased to 45 minutes.	Circulation time increased due to decreased flow rate. See Item 202.
B 3/4 1-2	Minimum Temp. for Criticality (Bases)	B	*	203	Revised discussion of interaction between minimum temperature for criticality requirement and P-12 reset point; paragraph reworded for consistency with Unit 2. Technical Specifications.	Bases were revised to more accurately reflect the operation of P-12 reset point.
B 3/4 1-2; B 3/4 1-3	3/4.1.2 (Bases)	B		204	"above" is changed to "below".	Editorial change; typographical error correction.
		B	*	205	Revisions were made to description of the RWST and BAST as boration sources.	Boration source volumes were adjusted to address dilution transient when operating on RHR at beginning of cycle. The higher boron concentration of the RWST is also reflected in the basis. Volumes used in the Technical Specifications which bound Units 1 & 2 are discussed. See Attachment 13.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		B		206	Mathematical symbols are written out in words.	Editorial change for clarity.
		B		207	pH value limits are added.	Limits reflect the analysis in Attachment 13.
B 3/4 2-1	3/4.2	B		208	"$F_Q(Z,\ell)$" is changed to "$F_Q(Z)$".	This change is based on the Exxon analysis presented in XN-NF-85-115(P), Rev. 2. This report was transmitted to the NRC with a letter dated January 15, 1987 from Exxon Nuclear Company, Inc. The Exxon letter was identified as GNW:001:87. This report was placed on our docket by a letter dated January 29, 1987 from M. P. Alexich to the NRC Document Control Desk. (Identifier AEP:NRC:0940E.) The new analysis does not result in a burnup dependence for Exxon fuel as discussed in Section 2.0 of XN-NF-85-115(P). This result is also discussed in a letter from H. G. Shaw to R. Bennett dated January 26, 1987. The letter from Mr. Shaw is included as Attachment 15. To facilitate this review we are retransmitting AEP:NRC:0940E and a proprietary version only of XN-NF-85-115(P) with Attachment 15.
		B		209	Updated minimum DNBR limit.	Editorial change. Updated to value in FSAR Table 3.6.3-1.
B 3/4 2-2	3/4.2.1	B		210	The word "of" is added.	Editorial change; grammatical error correction.
		B		210a	"signifigance" is change to "significance".	Editorial change; typographical error correction
		B		211	"$F_Q(Z,\ell)$" is changed to "$F_Q(Z)$".	See remarks for Item 208.
		B		212	Description of burnup dependent F_Q envelope is removed.	See remarks for Item 208.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		B		212a	Period replaced by comma.	Editorial change.
B 3/4 2-3	Figure B 3/4 2-1	B		212b	Figure is redrawn.	Editorial change for clarity.
B 3/4 2-4	3/4.2.2 3/4.2.3	B		213	The words "nuclear enthalpy hot channel factor" changed to "nuclear enthalpy rise hot channel factor".	Editorial change; makes Technical Specifications for both units more similar.
		B	*	214	References to F_Q and $F_{\Delta H}$ expanded to include proposed APL Technical Specification.	Editorial change.
		B		215	(No change for this identifier).	
B 3/4 2-5	3/4.2.3	B		216	"physics tests" is changed to "PHYSICS TESTS".	Editorial change; physics tests is a defined term.
		B		217	Section on burnup dependent F_Q for Exxon fuel removed.	See remarks for Item 208.
B 3/4 2-6	3/4.2.5 (Bases)	B	*	218	Discussion of flow rate surveillances are included.	Surveillance requirements revised to add CHANNEL CALIBRATION and flow measurement. Monthly flow surveillance is removed as redundant to shiftly surveillance. Resulting surveillance requirements are consistent with Unit 2 Technical Specifications. See Attachment 6, Item Number 10.
	3/4.2.6 (Bases)	B	*	219	This section is changed to an Allowable Power Level (APL) Technical Specification.	The APDMS is not used. The plant will operate below APL.
B 3/4 3-3	3/4.3.3.6 (Bases)	B	*	220	This section is removed.	The APDMS is not used. The plant will operate below APL.
B 3/4 4-1	3/4.4.1 (Bases)	B	*	221	References to three loop operation are removed.	Three loop operation in Modes 1 and 2 will be prohibited.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		B		222	Updated minimum DNBR limit.	Editorial change. Updated to value in FSAR Table 3.6.3-1.
		B		223	P-8 is changed to 31% of RTP.	Conservative change to make Unit 1 Technical Specifications more like the Unit 2 Technical Specifications.
		B	*	224	Additional operable loops are required with control rods capable of withdrawal.	The Plant Transient Analysis requires these changes based on the uncontrolled control rod bank withdrawal from subcritical. The proposed specification conservatively requires 3 pumps for consistency with Unit 2. See letter from E.P. Rahe, Jr. to D. Eisenhut dated July 9, 1984. (Identifier NS-TA-84-003). To facilitate this review, we are transmitting this letter as Attachment 17.
B 3/4 4-1; B 3/4 4-2	3/4.4.2 3/4.4.3 (Bases)	B		225	Text is moved from page B 3/4 4-1 to page B 3/4 4-2.	Editorial change.
B 3/4 4-13	3/4.4.11 3/4.4.12 (Bases)	B	*	226	Periods are converted to slashes.	Editorial change.
B 3/4 5-3	3/4.5.5 (Bases)	B		227	pH value limits are changed.	Limits reflect the analysis in Attachment 13.
		B		227a	Discussion of the difference between the analysis value and Technical Specification value of the RWST temperature is added.	The minimum RWST temperature is conservatively increased to the value for the Unit 2 LOCA analysis. The Unit 1 analysis was performed with an RWST temperature of 70°F.
B 3/4 7-1	3/4.7.1.1 (Bases)	B	*	228	References to three loop operation are removed.	Three loop operation in Modes 1 and 2 will be prohibited.
		B	*	229	Reference to Table 3.7-2 is changed to Table 3.7-1. This basis is condensed to one page.	Editorial change; incorrect table reference.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
B 3/4 7-2	3/4.7.1 (Bases)	B	*	230	Variable definitions are moved to previous page.	Editorial change; improve readability.
B 3/4 9-1	3/4.9.1	B		230a	The basis section from STS is substituted for existing basis and is augmented with a discussion of the increase in boron concentration requirement to 2400 ppm.	The required concentration is conservatively increased to agree with the RWST concentration. The result is a substantial increase in the amount by which the core is shutdown during refueling.
	3/4.9.5 (Bases)	B		230b	"CORE ALTERNATIONS" is changed to "CORE ALTERATIONS".	Editorial change; typographical error correction.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
2-2	Figure 2.1-1	1		231	Curve for 2250 psia is added.	Editorial change. See letter dated July 31, 1986, ENC-AEP/0511, H.G.Shaw to D.H.Malin found in Attachment 10.
3/4 1-3	4.1.1.2	1		232	Change "greater than" to "greater than or equal to".	Editorial change. Makes Unit 1 and Unit 2 more consistent.
		1		232a	Mathematical symbols are written out in words.	Editorial change for clarity.
	3.1.1.2.b	1		232b	Period added.	Editorial change.
3/4 1-4	3.1.1.3 4.1.1.3	8		233	Flow rate requirement reduced to 2000 gpm.	An analysis was performed to reduce the required reactor coolant flow rate to 2000 gpm. See Attachment 5 for discussion of heat removal, mixing, and stratisfication considerations. See Attachment 11 for dilution transient considerations.
		1		234	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 1-5; 3/4 1-6; 3/4 1-6a	3.1.1.4 Figure 3.1-2	8		235	The upper limit on MTC for operation above 70% RTP is changed. The upper limit is now graphically displayed (see Item 238).	To improve operational flexibility. Justification provided in Attachment 10.
	4.1.1.4.b	1		236	Specified 300 ppm surveillance at "RATED THERMAL POWER equilibrium boron concentration".	Editorial change; change made to clarify the intent of the surveillance requirement.
		1		237	Mathematical symbols are written out in words.	Editorial change for clarity.
		1		238	The new MTC limits proposed in Item 235 are now graphically displayed in Figure 3.1-2 on new page 3/4 1-6a.	Editorial change.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 1-8	3.1.2.1	4		239	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
	4.1.2.1.a	1		240	"≥ 145" is changed to "greater than or equal to 145°F".	Editorial change.
3/4 1-11	3.1.2.3	1		241	"the" is removed from footnote.	Editorial change; typographical error correction.
		1		241a	"ar" is changed to "are".	Editorial change; typographical error correction.
		1		242	Mathematical symbols are written out in words.	Editorial changes for clarity.
3/4 1-15	3.1.2.7.a.1	1		243	"of" is added. Word "contained" is removed.	Editorial changes; typographical error correction; clarification of meaning.
	3.1.2.7.b.1	1		243a	Word "contained" is removed.	Editorial change; clarification of meaning.
	3.1.2.7.b.2	8		244	RWST minimum boron concentration is changed.	The minimum RWST boron concentration limit has been increased to provide additional margin for the LOCA long-term cooling criterion. See Attachment 13. Attachment 13 includes the Westinghouse analysis and the evaluations of impacts on Unit 2 performed by Advanced Nuclear Fuels (Exxon) and AEPSC.
	3.1.2.7.b.3	11		244a	The required RWST temperature is increased to 80°F.	The minimum RWST temperature is conservatively raised to the temperature required for operability as a safeguards system in modes 1, 2, 3 & 4. The value of 80°F from the Unit 2 LOCA analysis is conservatively chosen.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
	4.1.2.7.b	11		244b	The RWST temperature will be monitored regardless of outside air temperature.	This is a conservative increase in surveillance requirements.
	ACTION	4		245	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
	3.1.2.8.a.1	11		246	Changed BAST minimum volume. Substituted "usable" for contained.	Boration sources are being changed to select the most conservative volume from the Unit 1 and Unit 2 analyses. For this value the Unit 1 analysis is more conservative.
	3.1.2.8.b.1	1		246a	Upper volume limit on RWST is removed.	The upper limit of 420,000 gallons is the capacity of the tank. The limit has no effect.
3/4 1-16	3.1.2.8.b.2	8		247	RWST minimum boron concentration is changed.	See remarks for item 244.
		8		248	RWST boron concentration upper limit is changed.	The revised containment sump pH analysis and the changeover to hot-leg recirculation safeguards analysis require a new upper limit on the RWST concentration. See Attachment 13.
3/4 1-17	4.1.2.7.b	11		248a	The RWST temperature will be monitored regardless of outside air temperature.	This is a conservative increase in surveillance requirements.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 1-18	3.1.3.1 ACTION c.1 ACTION c.2	1		249	"The rod" is changed to "The affected rod".	Editorial change for clarity.
3/4 1-18; 3/4 1-19	ACTION c.2.b	1		250	ACTION c.2.b is moved from page 3/4 1-19 to page 3/4 1-18.	Editorial change.
	ACTION c.2	1		251	Words added to emphasize that when ACTION c.2 is chosen that items a, b and c plus the choice between items d and e must be performed.	Editorial change.
3/4 1-19	ACTION c.2.d	1		252	Mathematical symbols are written out in words.	Editorial change for clarity.
	ACTION c.2.e	1		253	Reference to Figure 3.1-2 is removed.	Editorial change; three loop operation in Modes 1 and 2 was removed for Unit 2 in Amendment No. 82.
	4.1.3.1.1	1		254	References to part length rods are removed.	Editorial change; part length rods are not used.
	4.1.3.1.2	1		255	The words "in the core" are removed.	Editorial change. Makes Specifications of both units more similar.
3/4 1-23	3.1.3.4	1		256	"(228 steps)" is added.	Editorial change; clarifies meaning of fully withdrawn.
		1		257	Mathematical symbols are written out in words.	Editorial change for clarity.
	4.1.3.4	3		258	words "prior to entering Mode 2" replace "prior to reactor criticality."	Requiring the completion of this test prior to entering MODE 2 is conservative to requiring the test prior to criticality. MODE 2 is entered with the reactor subcritical by 1%. However, making the requirement mode dependent eases administrative control.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 1-24	3.1.3.5	1		259	"(228 steps)" is added.	Editorial change; clarifies meaning of fully withdrawn.
		1		260	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 1-25	3.1.3.6	1		261	"figures" is changed to "figure".	Editorial change.
		1		262	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 1-27		1		263	Page is removed.	Editorial change. Blank page not necessary at end of section.
3/4 2-1		1		264	APL footnote is removed.	Editorial change; APL is a defined term.
3/4 2-4	Figure 3.2-1	1		265	Figure is redrawn.	Editorial change for clarity.
3/4 2-16	Table 3.2-1	8		266	Footnote added to document flow allowance for measurement error. Analysis value reduced by the value of the allowance.	Omitted from letter to H. R. Denton from M. P. Alexich dated March 14, 1986 (Identifier AEP:NRC:0916I). To facilitate this review we are re-transmitting Attachment 7 of AEP:NRC:0916I as Attachment 12 to this letter. See page 2 of Attachment 12.
	Table 3.2-1	1		267	Footnote *** is added.	See Remarks for Item 266.
		1		267a	Asterisks moved to right hand column.	Editorial change.
	Footnote **	8		268	The words "at least three" are added.	This change reflects an analysis previously submitted in Attachment 3 to AEP:NRC:0916I for RCS Tavg and Attachment 7 of AEP:NRC:0916I for the pressurizer pressure. To facilitate this.review, Attachments 3 and 7 to AEP:NRC:0916I are retransmitted as Attachment 18 and 12, respectively, of this letter. See page (vii) of Attachment 18 and page 3 of Attachment 12.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
3/4 2-18	Table 3.2-2	8		269	Allowance for readability included for RCS Tavg and Pressurizer Pressure. The allowance was calculated consistently with footnote *.	Omitted from letter to H.R. Denton from M.P. Alexich dated March 27, 1986 (Identifier AEP:NRC:0916P). See Attachment 3 to AEP:NRC:0916I for RCS Tavg and Attachment 7 of AEP:NRC:0916I for the pressurizer pressure. To facilitate this review, Attachments 3 and 7 to AEP:NRC:0916I are retransmitted as Attachment 18 and 12, respectively, of this letter. See page (vii) of Attachment 18 and page 3 of Attachment 12.
3/4 2-19	3.2.6	1		270	ALLOWABLE POWER LEVEL is capitalized	Editorial change; ALLOWABLE POWER LEVEL (APL) is a defined term.
		1		271	Expression for APL is revised to more accurately reflect the meaning of APL.	Editorial change; APL cannot be greater than 100% of Rated Thermal Power.
		1		272	Second "$F_Q(Z)$" is replaced by "measured hot channel factor".	Editorial change for clarity.
3/4 2-19; 3/4 2-20		1		273	ACTION statements are moved from page 3/4 2-19 to page 3/4 2-20.	Editorial change.
3/4 3-3	Table 3.3-1 Items 13 & 14	1		274	"in" is added.	Editorial change; typographical error correction.
3/4 3-4	Table 3.3-1 Items 21 & 22	1		275	Clarifications made to properly identify which ACTION statements apply to each applicable mode.	Editorial change; this change corrects a format error made in the issuance of Amendment No. 86.
3/4 3-12	Table 4.3-1 Item 13	7		276	Loss of Flow-Two Loops Functional Unit has an added exemption from Specification 4.0.4.	This was omitted from letter from M. P. Alexich to H. R. Denton dated March 27, 1986

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
						(Identifier AEP:NRC:0916P). Exemption is provided for surveillance which must be performed in the applicable mode. The change was approved for "Loss of Flow - Single Loop" in Amendment 82 to DPR-74.
3/4 3-28	Table 3.3-5 Item 8a	1		277	Reactor trip is removed from description.	Editorial change; to make the proposed Technical Specifications between Units more similar. The response time for ESF Steam Generator Water Level-High High turbine trip is not modeled in the current analysis of record.
3/4 4-2	3.4.1.2.d	11		277a	Criterion for the operability of reactor coolant loops based on P-12 is added.	Table 3.3-3 requires at least three loops operating above P-12. This ensures flow through RTD by-pass loops. This provision is added for consistency with Table 3.3-3. An appropriate ACTION statement has been proposed to correspond to this requirement.
	ACTION b ACTION c	11		277b	ACTION statements added to address too few reactor coolant loops when control rods are capable of withdrawal. Old ACTION b becomes ACTION d.	Proposed to maintain similarity to Unit 1. See Item 135.
3/4 4-2a	ACTION d	1		277c	ACTION d and footnotes moved from previous page.	Editorial change; additional text requires moving this material.
3/4 4-3	3.4.1.3 ACTION b	11		277d	ACTION statement added to address too few reactor coolant loops when control rods are capable of withdrawal. Old ACTION b becomes ACTION c.	Proposed to maintain similarity to Unit 1. See Item 135.
3/4 4-4	3.4.2	4		278	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected

PAGE	SECTION	+ *	#	DESCRIPTION	REMARKS
					operating conditions.
3/4 5-1	3.5.1.c	8	279	Minimum accumulator boron concentration is change.	The minimum accumulator boron concentration limit has been increased to provide additional margin for the LOCA long-term cooling criterion. See Attachment 13.
	3.5.1.c	8	280	Maximum accumulator boron concentration is changed.	The revised containment sump pH analysis and changeover to hot-leg recirculation safeguards analysis establish a new upper limit on accumulator boron concentration. See Attachment 13.
	3.5.5.a	1	280a	Upper volume limit on RWST is removed.	The upper limit of 420,000 gallons is the capacity of the tank. The limit has no effect.
3/4 5-11	3.5.5.b	8	281	Minimum RWST boron concentration is changed.	The minimum RWST boron concentration limit has been increased to provide additional margin for the LOCA long-term cooling criterion. See Attachment 13.
		8	282	Maximum RWST boron concentration is changed.	The revised containment sump pH analysis and changeover to hot-leg recirculation safeguard analysis establish a new upper limit on RWST boron concentration. See Attachment 13.
	4.5.5.b	11	282a	The RWST temperature will be monitored regardless of outside air temperature.	This is a conservative increase in surveillance requirements.
3/4 8-5	3.8.1.2	4	283	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		1		284	Existing footnote * is changed to footnote **.	Editorial change.
3/4 9-1	3.9.1	1		285	Mathematical symbols are written out in words.	Editorial change for clarity.
	3.9.1.b ACTION	11		285a	The required boron concentration for refueling is increased to 2400 ppm.	The required concentration is conservatively increased to agree with the RWST concentration. The result is a substantial increase in the amount by which the core is shutdown during refueling.
		4		286	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
3/4 9-2	3.9.2	4		287	Footnote added.	The Technical Specification boron concentration in the RWST is sufficient to provide adequate shutdown margin from expected operating conditions.
		1		288	Footnote removed.	Editorial change; the 1984 Refueling Outage has been completed.
3/4 9-8	3.9.8.1	8		289	Flow rate requirement reduced to 2000 gpm.	An analysis was performed to reduce the required reactor coolant flow rate to 2000 gpm. See Attachment 5 for discussion of heat removal, mixing, and stratisfications considerations. See Attachment 11 for dilution transient considerations.
		1		290	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 10-3	3.10.3.b	12		291	Specification is reworded.	Clarifies intention of specification. See letter from Westinghouse found in Attachment 6, Item Number 14.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
		1		292	"reactor trip setpoints" is changed to "Reactor Trip Set- points".	Editorial change for consistency with 3.10.4.b.
	3.10.3.b ACTION 4.10.3.1	1		293	Mathematical symbols are written out in words.	Editorial change for clarity.
3/4 10-4	3.10.4.b	12		294	Specification is reworded.	Clarifies intention of specifica- tion. See letter from Westinghouse found in Attachment 6, Item Number 14.
	4.10.4.1	1		294a	"the" is added.	Editorial change.
		1		295	Mathematical symbols are written out in words.	Editorial change for clarity.
6-19	6.9.2.h	1		296	Moderator Temperature Coefficient is added to the Special Reports list.	Editorial change; A Special Report is to be submitted to the NRC within 10 days of exceeding the limit of Figure 3.1-2.
	6.9.2.e	1		296a	Comma is removed.	Editorial change; typographical error correction.
B 3/4 1-3	3/4.1.2 (Bases)	B		297	Revisions made to the description of the RWST as a boration source.	The higher boron concentration of the RWST is reflected in the basis. Volumes used in the Technical Specifica- tions which bound Units 1 and 2 are discussed. See Attachment 13.
		B		298	pH value limits are changed.	Limits reflect the analysis in Attachment 13.
B 3/4 4-1a B 3/4 4-2	3/4.4.2 3/4.4.3 3/4.4.4 (Bases)	B		299	Text is combined to one page; B 3/4 4-1a is to be removed.	Editorial change; removes dupli- cation of text that was included with License Amendment No. 82.

PAGE	SECTION	+	*	#	DESCRIPTION	REMARKS
B 3/4 5-3	3/4.5.5 (Bases)	B		300	pH value limits are changed.	Limits reflect the analysis in Attachment 13.
B 3/4 9-1	3/4.9.1	B		301	The basis section from STS is substituted for existing basis and is augmented with a discussion of the increase in boron concentration requirement to 2400 ppm.	The required concentration is conservatively increased to agree with the RWST concentration. The result is a substantial increase in the amount by which the core is shutdown during refueling.

Attachment 5 TO AEP:NRC:0916W

ANALYSIS OF HEAT REMOVAL AT

2000 GPM PRIMARY FLOW AND

EVALUATION OF MIXING AND STRATIFICATION

T/S 3/4.1.1.3 (Reactivity Control Systems - Boron Dilution) presently requires an RCS flow rate of 3000 gpm whenever a reduction in RCS boron concentration is being made. As discussed in the Bases for this T/S, the purpose of this requirement is to provide adequate mixing, prevent boron stratification, and ensure that reactivity changes will be gradual during boron concentration reductions in the RCS. Similarly, T/S 3/4.9.8.1 (Refueling Operations - Residual Heat Removal and Coolant Circulation) requires 3000 gpm of RHR flow during Mode 6 operation. According to the Bases for this T/S, its purpose is to (1) ensure sufficient cooling capacity is available to remove decay heat and (2) maintain sufficient coolant circulation through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification. In practice, however, the 3000 gpm requirement can present severe operational difficulties because of the possibility of pump vortexing. This concern exists during RHR system operation in Mode 5 (cold shutdown), when the RCS may be partially drained down to facilitate various maintenance operations (half-loop operation). Problems with vortexing of the RHR pumps during half-loop operation has been a recurring problem in the industry, and was the subject of IE Information Notice No. 86-101. Because loss of the RHR pumps due to vortexing could conceivably cause the loss of decay heat removal capability, we have performed analyses to demonstrate that 2000 gpm flow is sufficient for the purposes of T/Ss 3/4.1.1.3 and 3/4.9.8.1. Analyses addressing the boron dilution incident concerns are discussed in Attachment 14 for Unit 1 and in Attachment 11 for Unit 2. Analyses for mixing and boron stratification concerns and decay heat removal capability are discussed below. References for these analyses and a nomenclature list are included at the end of this attachment.

1. Mixing and Boron Stratification

Summary

Boron stratification was assessed by comparing the turbulence and core crossflow that would exist at 2000 gpm to what exists at 3000 gpm. At 3000 gpm (the current technical specifications limit) boron stratification would not occur. The evaluation showed that the Reynolds number in all RCS piping, RHR piping and the reactor vessel downcomer is in the turbulent region. Turbulence in the downcomer would promote mixing, thereby reducing any concentration gradients that may have existed when the fluid entered the downcomer. Upon entering the lower plenum of the reactor vessel, the momentum of the fluid combined with the effects of a sudden expansion would tend to entrain surrounding fluid, further reducing concentration gradients. Finally, crossflow in the core would promote additional mixing. The crossflow is a function of the Reynolds number to the 0.9 power (References 2 and 3). The crossflow at 2000 gpm would be 69% of that at 3000 gpm. Thus, a significant amount of crossflow would exist. (See analysis section.)

Once the flow exits the core, the RHR piping turbulence would be very high, and considerable mixing would continue, especially as the fluid flows through the pump.

Based on the forgoing mixing evaluation, it is concluded that boron stratification is no more of a concern with 2000 gpm RHR flow than with 3000 gpm RHR flow.

Details of Analysis

There are several places in the RCS where mixing could occur. These include the reactor coolant system piping, the reactor vessel downcomer, the reactor vessel upper and lower plenum, the core region, and the RHR system piping. For all of these regions except the plenums, Reynolds numbers were calculated. (Although a calculation was not made for the plenums, the core area was determined to be the least turbulent flow region and thus would bound the plenums.) These Reynolds numbers are listed below.

Reynolds Numbers

Location	Reynolds Number	
	3000 gpm	2000 gpm
Reactor Inlet	151,600.	101,000.
Reactor Outlet	144,100.	96,000.
Reactor Downcomer	49,400.	33,100.
Reactor Core	840.	560.
RHR Piping	1,271,000.	847,000.

As can be seen in the table above, all areas of interest except the core had Reynolds numbers well in excess of 4000, at 2000 gpm flow, and thus it was concluded that the flow in these regions would be turbulent and that adequate mixing would occur.

In the core and plenum regions, flow is laminar. However, there is mixing due to crossflow within the core region. A mixing parameter

$$B = G/\bar{G} = K \, Re^{-0.1}$$

exists which ratios cross flow in the core to the average core flow (Ref. 2). Since the Reynolds number is directly proportional to the flow in the system, the equation can be modified to give

$$G = K'(\bar{G})^{0.9}$$

From this, the crossflow of two different flow rates can be compared by using:

$$G_2/G_1 = [\bar{G}_2/\bar{G}_1]^{0.9}$$

Using this equation, it can be seen that the crossflow at 2000 gpm RHR flow compared to 3000 gpm would be $(2000/3000)^{0.9}$, or approximately 69% of the value at 3000 gpm. This remaining crossflow, together with the mixing that would occur in the piping and the downcomer, is judged to be sufficient to prevent significant boron stratification.

2. Decay Heat Removal

Summary

Calculations were performed to determine the minimum RHR flow which would be required to remove decay heat. The assumptions used in the analysis were that the lake water temperature is 85°F and the maximum temperature of the reactor coolant water is 200°F. The lake serves as the ultimate heat sink for the decay heat generated in the core. This temperature ultimately determines the required coolant flow to the reactor core. The maximum reactor coolant temperature is set by the technical specification limit of 200°F in Mode 5, cold shutdown. To account for uncertainties, additional calculations were performed with a margin of 20% added to the decay heat value and the lake temperature increased to 95°F.

It was also assumed that the product of overall heat transfer coefficient and surface area (UA) was constant and equal to the design value in the CCW heat exchanger. A ratio was computed for UA as a function of reduced flow for the RHR heat exchanger. Flow rates other than RHR flow rate (such as CCW and ESW loops) were also assumed to remain constant and equal to design values. Constant pressure specific heat was taken as 1.0 Btu/lbm/°F for all flow streams. Decay heat was calculated using Reference 1 methodology.

These calculations demonstrated that RHR flow of 2000 gpm would be more than sufficient to remove decay heat, even with the reactor drained to the half-loop condition.

Details of Analysis

This section provides details of calculations to determine the minimum flow rate required to remove the decay heat from the reactor. The RHR system was modeled using the flow diagram shown in Figure 1. The problem involves six equations and six unknowns (the temperature of each stream). The basic equations to be solved are:

(1) $Q_i = m_i \, C_p \, \Delta T_i$

and

(2) $Q_i = (UA)_i \Delta T_{LM,i} F_i$

Equation (1) describes the sensible heat gain or loss in the coolant. Equation (2) describes the heat transfer between the fluids flowing on the shell side and the tube side of the heat exchanger. The log mean temperature difference, ΔT_{LM} in equation 2, compensates for the fact that the temperature difference between the hot and cold fluid may change as both fluids traverse the heat exchanger.

The product of the overall heat transfer coefficient, U, and the heat exchanger surface area, A, was determined from the design condition given in Table 1. This was accomplished by rearranging equation 2 to give

$$(3)\ UA = \frac{Q}{(\Delta T_{LM})(F)}$$

The calculated values of UA are given in Table 1.

The decay heat used in the equation was determined by

$$(4)\ P/P_o\ (t_o, t_s) = P/P_o\ (\infty, t_s) - P/P_o\ (\infty,\ t_o + t_s)$$

where

P/P_o = Power to full-power ratio

t_o = Effective full power seconds at 3411 MW

t_s = Number of seconds since shutdown

$$(5)\quad P/P_o\ (\infty,\ t_s) = A\ t_s^{-a}$$

Where A and a are values obtained from Reference 1.

Based on 1202 effective full-power days (Reference 5), the decay heats were calculated for decay times of 2.5 to 6.0 days. These results are given in Table 2.

The component cooling water has heat loads other than the decay heat from the core. The total amount of these heat loads was obtained from the design values and was found to be 34.9 (10^6) Btu/hr. For the calculation of the minimum low flow, the decay heat at 2.5 days, 40.4 (10^6) Btu/hr, was used. This made the total heat load 75.3 (10^6) Btu/hr.

Mass flows in the system, (other than RHR flow which will be calculated), were obtained from the design values. The component cooling water flow which is diverted to the auxiliaries is summarized in Table 3. The mass flows used in the calculation are summarized in Table 4.

The minimum mass flow rate required to remove decay heat after 2.5 days with a lake temperature of 85°F was determined by iteration to be approximately 1000 gpm.

To account for uncertainties in the decay heat value, a margin of 20% was conservatively added, the lake temperature was conservatively increased to 95°F and the calculation repeated. When this was done, the minimum required flow was determined to be approximately 1450 gpm.

TABLE 1

Heat Exchanger Design Conditions

RHR Heat Exchanger

Design Heat Load, Btu/hr	41.1×10^6
Shell Side Inlet Temperature, ^oF	95.
Tube Side Inlet Temperature, ^oF	140.
Shell Side Outlet Temperature, ^oF	111.6
Tube Side Outlet Temperature, ^oF	112.3
Calculated UA, Btu/hr ^oF	1.836×10^6

CCW Heat Exchanger

Design Heat Load, Btu/hr	76×10^6
Shell Side Inlet Temperature, ^oF	114.
Tube Side Inlet Temperature, ^oF	76.
Shell Side Outlet Temperature, ^oF	95.
Tube Side Outlet Temperature, ^oF	92.
Calculated UA, Btu/hr ^oF	3.71×10^6

References 5, 7

TABLE 2

Decay Heat as a Function of Time

Time after Shutdown, Days	Decay Heat, 10^6 Btu/hr
2.5	40.4
3.0	38.1
3.5	36.2
4.0	34.6
4.5	33.3
5.0	32.1
5.5	31.1
6.0	30.2

TABLE 3

CCW Auxiliary Cooling Water Flows

Component	Flow, gpm
Reactor Coolant Pump	560
Sealwater Heat Exchanger	38
Letdown Heat Exchanger	300
Spent Fuel Heat Exchanger	1500
RHR Pump	10
SI Pump	40
Spray Pump	20
Charging Pump	90
Penetrations	300
Gas Compressor	13
Reactor Support	40
TOTAL	2911

Reference 6

TABLE 4

Mass Flows Used In Analysis

Flow Stream (Refer to Figure 1)	Mass Flow, 10^6 lb/hr
M_1	To be calculated
M_2	2.56
M_3	4.67
M_4	4.0

References 4, 5, 6

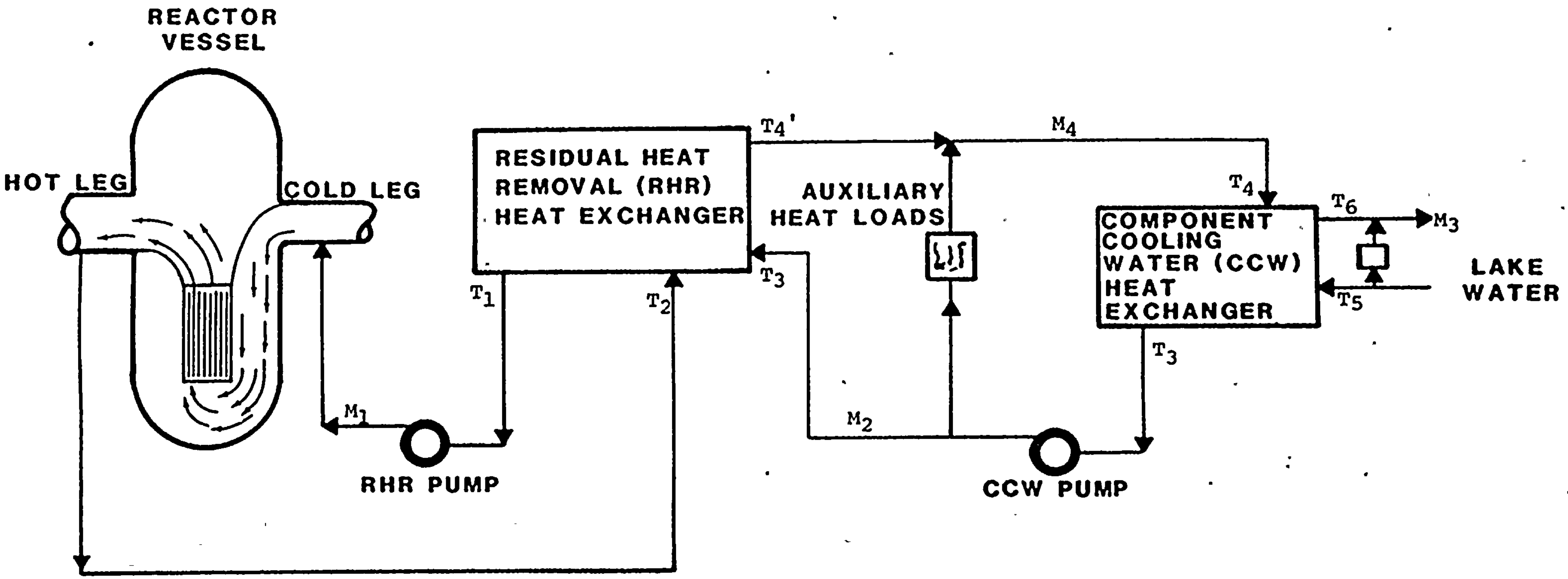


FIGURE 1. RHR LOOP CONFIGURATION

References

(1) NUREG/CR-2507 "Background and Derivation of ANS 5.4 Standard Fission Product Release Model," January, 1982.

(2) XN-NF-81-73, "Turbulent Mixing in Rod Bundles," October, 1982.

(3) McCabe and Smith, "Unit Operations of Chemical Engineering," 1956, p. 53.

(4) FSAR, Table 9.8-5.

(5) FSAR, Table 9.5-3.

(6) FSAR, Table 9.5-2.

(7) FSAR, Table 9.3-2.

NOMENCLATURE

Q_i	-	heat flow	Btu/hr
M_i	-	mass flow rate for stream i	lb/hr
C_p	-	heat capacity	Btu/lb ^oF
ΔT_i	-	temperature difference for stream i	^oF
U_i	-	overall heat transfer coefficient for heat exhanger i	Btu/hr ft^2 ^oF
A_i	-	heat transfer area for heat exchanger i	ft^2
$\Delta T_{LM, i}$	-	log mean temperature difference for heat exchanger i	^oF
P_o	-	actual reactor power	MW
P	-	rated reactor power	MW
t	-	effective full-power operating time	Days
t_s	-	time since shutdown	Days
G	-	cross-flow	lb/ft^2 sec
$\overline{G}$	-	average coolant flow	lb/ft^2 sec
B	-	mixing parameter	dimensionless
F_i	-	configuration correction factor for heat exchanger i	dimensionless
K, K'	-	proportionality constants	

Attachment 6 TO AEP:NRC:0916W

EVALUATIONS OF PROPOSED TECHNICAL SPECIFICATIONS PERFORMED BY WESTINGHOUSE ELECTRIC CORPORATION

Westinghouse
Electric Corporation

Power Systems

Energy Systems
Service Division

Box 355
Pittsburgh Pennsylvania 15230-0355

AEP-87-135/REV. 2

March 6, 1987

Mr. E. G. Lewis
Nuclear Materials and Fuel Management Section
American Electric Power Service Corporation
One Riverside Plaza
Columbus, OH 43216

AMERICAN ELECTRIC POWER SERVICE CORPORATION
Response to D. C. Cook Unit 1 Tech Spec Changes

Dear Mr. Lewis:

The purpose of this letter is to revise our previous transmittal of the attached Technical Specification Changes. The revisions are minor in nature and involve items numbers 4 and 9.

In the "response" section of item number 9, reference to "Reference 4" was deleted. In item number 4, the Technical Specification originally supplied to Westinghouse relative to the MTC change was replaced by AEP with another Technical Specification page. At AEP's request, Westinghouse reviewed this change and found it acceptable. It is now a part of the attachments.

Please note these changes in the attachments to this letter. If you have questions, please do not hesitate to contact us.

Very truly yours,

H. C. Walls, Project Manager
Mid-America Area
U. S. Nuclear Projects

cc: J. G. Feinstein
E. G. Lewis
V. Vanderburg
J. M. Cleveland
W. G. Smith, Jr.

8981f/fc

ITEM NUMBER 1

1. AEP Comment: Review the revised basis (specification 2.2.1 reactor trip system instrumentation setpoints) for loss of flow. Verify that the revisions made by AEPSC in removing reference to 3-loop operations are consistent with Westinghouse methodology.

   Response: The reference to three loop operation may be deleted. In addition to what was deleted in the basis by AEP, the following statement (Page B 2-6) should also be deleted since it is applicable for three loop operation:

   "This latter trip will prevent the minimum value of the DNBR from going below the applicable safety analysis design limit DNBR value for each fuel type, (as listed in the bases for (Section 2.1.1) during normal and operational transients and anticipated transients when three loops are in operation and the overtemperature Delta-T trip setpoint is adjusted to the value specified for all loops in operation."

ITEM NUMBER 9

2. AEP Comment: Confirm which parameters in DNB specification (3.2.5, Table 3.2-1) have readability allowances. What is the accurate manner to address the error penalty in flow? (Analog of 3.5% penalty in standardized technical specifications). Confirm the treatment of measurement allowances in the draft DNB basis (3/4.2.5 DNB parameters) is correct.

   Response: The value for reactor coolant system T-avg (570.4 degrees F) was verified to include measurement uncertainties and is the indicated value as read in the control room. The T-avg indicator for at least three loops is read, added together, and divided by the number of loops measured (three or four), to obtain the reactor coolant system average temperature. It is recommended that the footnote in the proposed tech. specs. (Table 3.2-1), "indicated average of operable instrument loops" be changed to "indicated average of at least three operable instrument loops". The value for pressurizer pressure was verified to be the safety analysis bounding value.

   The value for reactor coolant system total flow rate (1.386 times ten to the eighth power pounds per hour) in Table 3.2-1, (DNB parameters) is an "indicated" value to which the flow rate must be compared to, to demonstrate compliance with this specification.

   It is acceptable to add the statement "the indicated values of T-avg and flow include allowances for instrument errors." To the basis of specification 3/4.2.5, DNB parameters. It is recommended that the last statement in the first paragraph be revised as follows: "Measurement uncertainties have been accounted for in determining the DNB parameters limit values.

ITEM NUMBER 10

3. AEP Comment: "Review new primary flow surveillance requirements (specification 3.2.5 DNB parameters and basis for 3/4.2.5). Monthly surveillance removed per discussion with R. Jansen in connection with Unit 2 T/S revisions.

Response: The monthly total flow rate surveillance (specification 4.2.5.2 in the current D. C. Cook Unit 1 tech specs) may be removed since the total flow rate is verified once every 12 hours. It is acceptable to add the surveillances on the channel calibration of the flow indicators and the total flow rate measurement. The revised basis for specification 3/4.2.5 adding the discussion on the new surveillances added, and the deletion of the discussion on the monthly flow surveillance is acceptable.

ITEM NUMBER 11

4. AEP Comment: Confirm that the augmented startup test program is completed.

Response: The augmented startup test program is complete and the proposed tech spec change in specification 3.10.2 may be implemented.

ITEM NUMBER 4

Provide justification for changing MTC from a step to a ramp function of power as proposed by Westinghouse.(18)

Response: A safety evaluation of the proposed change to the D. C. Cook Unit 1 moderator temperature coefficient (MTC) Technical Specification 3/4.1.1.4, has been completed. Specifically, American Electric Power has expressed an interest in changing the form of the MTC spec from a "step" of +5 to 0 pcm/^{o}F at 70% power to a "ramp" of +5 pcm/^{o}F at 70% power to 0 pcm/^{o}F at 100% power.

The following accidents, determined to be sensitive to a positive MTC, were analyzed in support of the OFA transition for the Cycle 8 reload transition:

RCCA Bank Withdrawal from Subcritical

RCCA Bank Withdrawal at Power

Loss of Reactor Coolant Flow (including Locked Rotor Analysis)

Loss of External Load

Excessive Heat Removal Due to Feedwater System Malfunction

RCCA Ejection

With two exceptions, the current safety analyses were based on a +5 pcm/^{o}F MTC, which was assumed to remain constant for variations in temperature. The assumption of a +5 pcm/^{o}F MTC existing at full power is conservative, since the proposed Technical Specification requires that the coefficient be linearly ramped to zero above 70% power. Therefore, the conclusions presented in the cycle 8 reload transition safety analyses (the current analyses) remain valid.

The RCCA ejection and RCCA Bank Withdrawal from Subcritical analyses performed in support of the Cycle 8 reload transition were based on a coefficient which was at least +5 pcm/^{o}F at the appropriate zero or full power nominal average temperature, and which became less positive for higher temperatures. This was necessary since the TWINKLE computer code used in the analyses is a diffusion-theory code rather than a point-kinetics approximation and the moderator temperature feedback cannot be artificially held constant with temperature. The conclusions of the Cycle 8 reload transition analyses remain valid.

Since this proposed Tech Spec change does not alter the previous Tech Spec limits for MTC at 0% power and at 100% power, the results of the large break and small break LOCA analyses and long term core cooling calculation will not be affected by this change.

A copy of the proposed D. C. Cook Unit 1 Technical Specification 3/4.1.1.4 is attached, incorporating the MTC change. A Nuclear Safety Evaluation Checklist has been completed for this evaluation and is attached.

ITEM NUMBER 14

5. AEP Comment: Documentation may be needed that states that the high flux low setpoint is sufficient during physics tests. (Specifications 3.10.4 physics tests and 3.10.5 natural circulation tests) we have received the interpretation verbally from R. Jansen on Westinghouse.

   Response: Westinghouse recommends that 3.10.4 B. and 3.10.5 B. be changed to read as follows:

   > The reactor trip setpoints for the operable intermediate range, neutron flux and the power range, neutron flux, low setpoint are set at less than or equal to 25% of rated thermal power.
   >
   > The justification for this change is for clarification purposes, the intent of the spec is not changed.

ATTACHMENT I

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Within the region of acceptable operation in Figure 3.1-2, and

b. Less negative than -3.5×10^{-4} $\Delta k/k/^{\circ}F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

* With K_{eff} greater than or equal to 1.0

See Special Test Exception 3.10.4

1. The following discussion pertains to a discrepancy between a statement in the D. C. Cook Unit 1, Cycle 8 reload transition safety report Rupture of a Steam Pipe write-up and the plant's actual configuration.

Discussion

Appendix C.3.11 of the D. C. Cook Unit 1, Cycle 8 reload transition safety report discussed the analysis of a Rupture of a Steam Pipe. Condition E of the write-up stated that "Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location, would have the same effect on the NSSS as the 1.4 square foot break." In actuality, the steam generators for D. C. Cook Unit 1 are not equipped with integral flow restrictors. However, the reanalysis performed did assume the correct plant configuration. The most limiting steamline break scenario was assumed in the analysis. The case analyzed for a Rupture of a Steam Pipe was a complete severance of a pipe at the outlet of the steam generator (break area = 4.6 square feet) upstream of the flow restrictor, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available. As such, condition E of Appendix C.3.11 of the Cycle 8 reload transition safety report should be replaced with the following:

E. The most limiting case of a rupture of a steam pipe was analyzed. This was determined to be a break at the outlet of the steam generator (break area = 4.6 square feet) upstream of the flow restrictor, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available. This case has been considered in determining the core power and RCS transients.

ATTACHMENT II

WESTINGHOUSE PROPRIETARY CLASS 2

2. The following provides a discussion of a Feedwater System Malfunction transient assumption, regarding the modelling of reactor trip on turbine trip, for the Cycle 8 reanalysis.

Discussion

The current Feedwater System Malfunction analysis (presented in the Cycle 8 reload transition safety analysis report) was performed assuming a fully open feedwater control valve and is terminated by a steam generator hi-hi level trip signal which closes all FW control and isolation valves, trips the FW pumps, and trips the turbine. The feedwater system malfunction event is the only FSAR accident that assumes a turbine trip on steam generator hi-hi level. A reactor trip on turbine trip was then assumed to prevent reactor coolant heatup consistent with the cooldown characteristics of the feedwater malfunction event. The reactor trip on turbine trip was assumed as an anticipatory trip. If the reactor trip was not assumed, the transient would turn into a heatup event - in particular, a loss of normal feedwater due to the feedwater isolation which occurs on steam generator hi-hi level. A reactor trip would then be provided by a low-low steam generator water level signal. Further, the reactor trip on turbine trip is not required for core protection for the feedwater malfunction event. The results (minimum DNBR) of the feedwater malfunction accident would be essentially unchanged if the reactor trip was not assumed to occur on turbine trip. Therefore, a reactor trip on turbine trip is not required in any non-LOCA transient for core protection.

SECL NO. NS-SECL-87-041

Customer Reference No(s)
B.O. No. CB10091
Westinghouse Reference No(s)
WAF No. A-13485

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) D. C. Cook Unit 1

2) CHECK LIST APPLICABLE TO: Change to MTC Tech Spec
(Subject ofChange)

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

(3.1) Yes No X A change to the plant as described in the FSAR?
(3.2) Yes No X A change to procedures as described in the FSAR?
(3.3) Yes No X A test or experiment not described in the FSAR?
(3.4) Yes X No A change to the plant technical specifications (Appendix A to the Operating License)?

4) CHECK LIST - PART B (Justification for Part B answers must be included on Page 2.)

(4.1) Yes No X Will the probability of an accident previously evaluated in the FSAR be increased?
(4.2) Yes No X Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes No X May the probability of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR b increased?
(4.5) Yes No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR b increased?
(4.6) Yes No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(4.7) Yes No X Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in 4) cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to NRC pursuant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification upon the written safety evaluation, (1)for answers given in Part B of the Safety Evaluation Check List:

An evaluation has determined that the current safety analyses for D. C. Cook Unit 1 support a Tech Spec change for the moderator temperature coefficient (spec 3/4.1.1.4). The spec will be changed from a "step" of +5 to 0 pcm/°F at 70% power to a "ramp" of +5 pcm/°F at 70% power to 0 pcm/°F at 100% power.

(1) Reference to document(s) containing written safety evaluation:

NS-OPLS-TA-II-87-028

FOR FSAR UPDATE

Section: Page(s): Table(s): Figure(s):

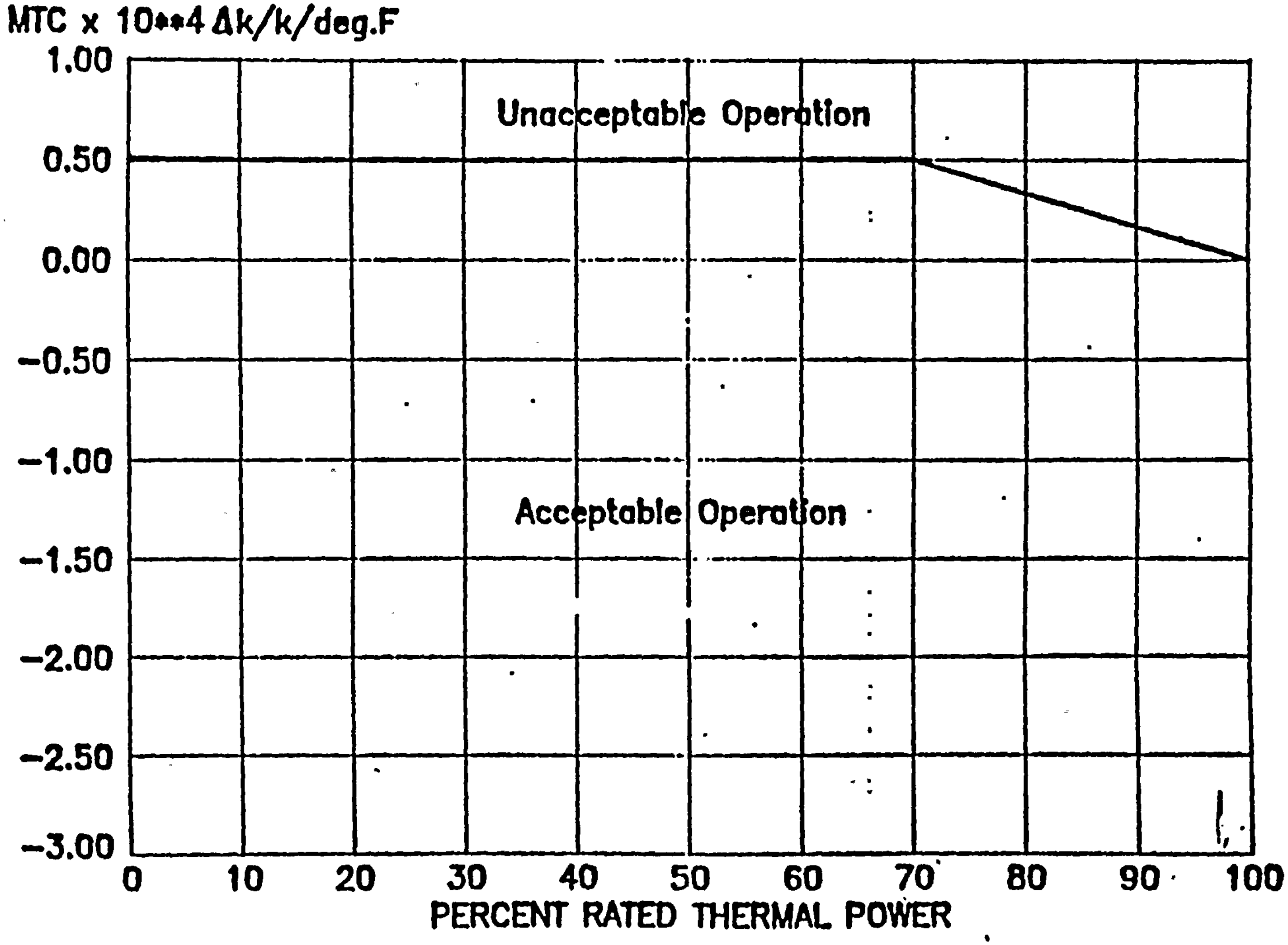
Reason for/Description of Change:

6) APPROVAL LADDER

6.1) Prepared by (Nuclear Safety): Glenn Hellerle Date: 1/22/87
6.2) Coordinated with Engineer(s): Date: 1/20/87
6.3) Coordinating Group Manager(s): Date: 1/20/87
6.4) Nuclear Safety Group Manager: Date: 1-23

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



Attachment 7 TO AEP:NRC:0916W

ROD INSERTION LIMIT INTERCEPTS

SUPPLIED BY WESTINGHOUSE ELECTRIC CORPORATION

MAR 5 1986

Westinghouse
Electric Corporation

Water Reactor
Divisions

Nuclear Fuel Division

Box 3912
Pittsburgh Pennsylvania 15230-3912

February 27, 1986

S6AE*-G-0019

W-AEP/0243

Keywords: AEP RIL
Tech-Specs

Indiana and Michigan Electric Co.
c/o Joseph L. Bell
Engineer, Nuclear Materials and Fuel
Management
American Electric Power Service Corp.
One Riverside Plaza, 20th Floor
Columbus, OH 43215

Dear Mr. Bell:

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D.C. COOK UNIT 1
TECHNICAL SPECIFICATIONS ROD INSERTION LIMITS

Attached are change pages to be incorporated in the D.C. Cook Unit 1 Technical Specifications. The RIL limit lines being submitted here for 3-loop and 4-loop operation are no different from the ones in the current Tech Specs.

At your request, Westinghouse is incorporating on those limit lines the actual endpoints in steps withdrawn at both HZP and HFP for control banks D and C.

If you have any questions, please call.

Very truly yours,

J.C. Miller
Project Engineer
NFD Fuel Projects

cc: M.P. Alexich
J.M. Cleveland
D.H. Malin - w/enc.
V.D. Vanderburg
W.L. Zimmermann

WESTINGHOUSE PROPRIETARY CLASS 2

CDC-86-058

February 26, 1986

F. J. Silva, 412/374-2189
Westinghouse Nuclear Fuel Division
Core Engineering
MMOB-301 MS 3-28
P. O. Box 3912, Pittsburgh, Pa. 15230

MEMO TO: J. C. Miller
NFD Fuel Projects

cc: B. M. Bowman
B. J. Johansen

SUBJECT: D. C. Cook Unit 1 Tech Specs Rod Insertion Limits

KEYWORDS: AEP TECH-SPECS RIL

Attached please find change pages to be incorporated in the D. C. Cook Unit 1 Technical Specifications. The RIL limit lines being submitted here for 3-Loop and 4-Loop operation are no different from the ones in the current Tech Specs. At AEP request we are incorporating on those limit lines the actual endpoints in steps withdrawn at both HZP and HFP for control banks D and C.

Please send this information to AEP to be submitted to the NRC together with other Tech Specs changes.

F. J. Silva

F. J. Silva
CE Core Design C

W L Orr

APPROVED: W. L. Orr, Manager
CE Core Design C

FJS:

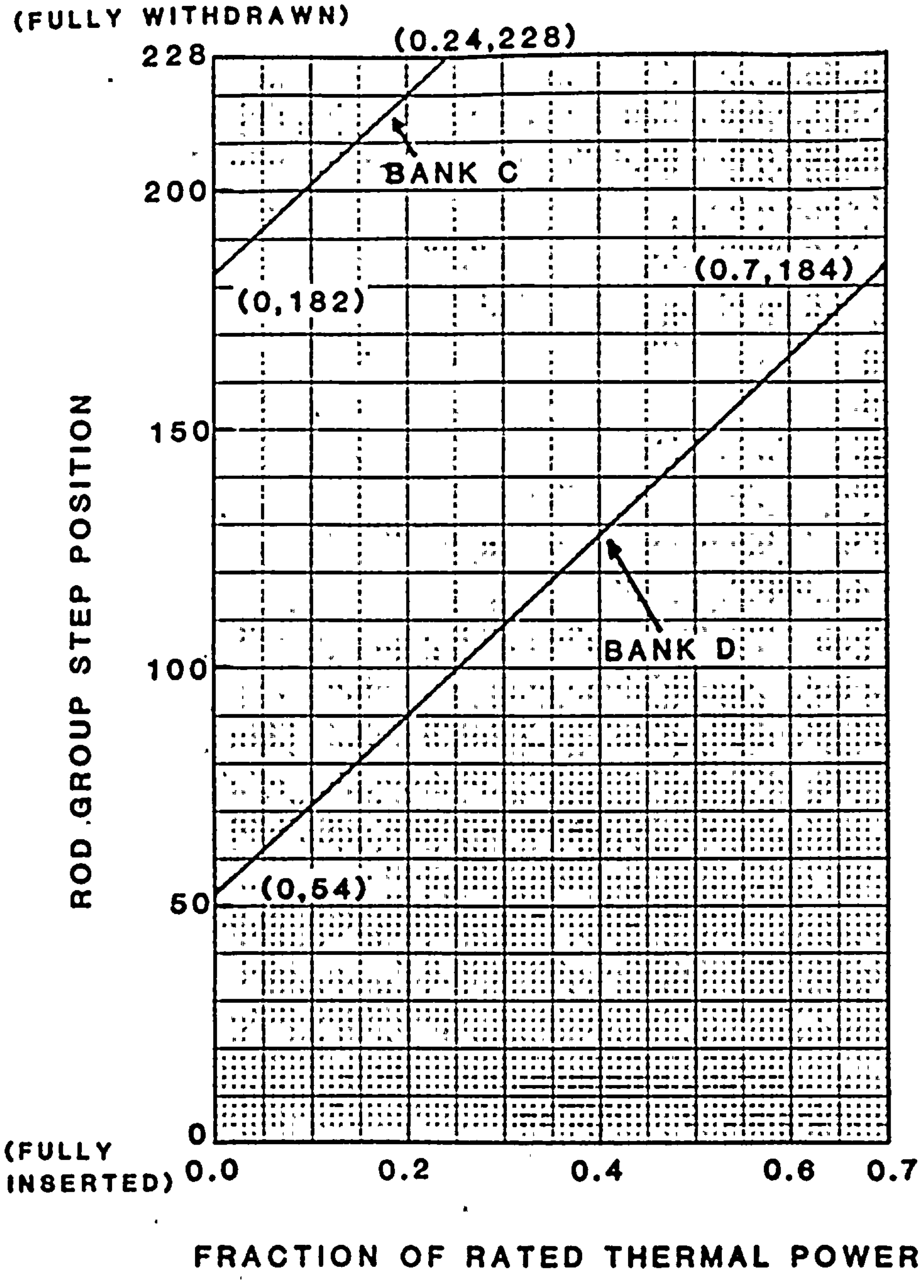


FIGURE 3.1-1 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER 3 LOOP OPERATION

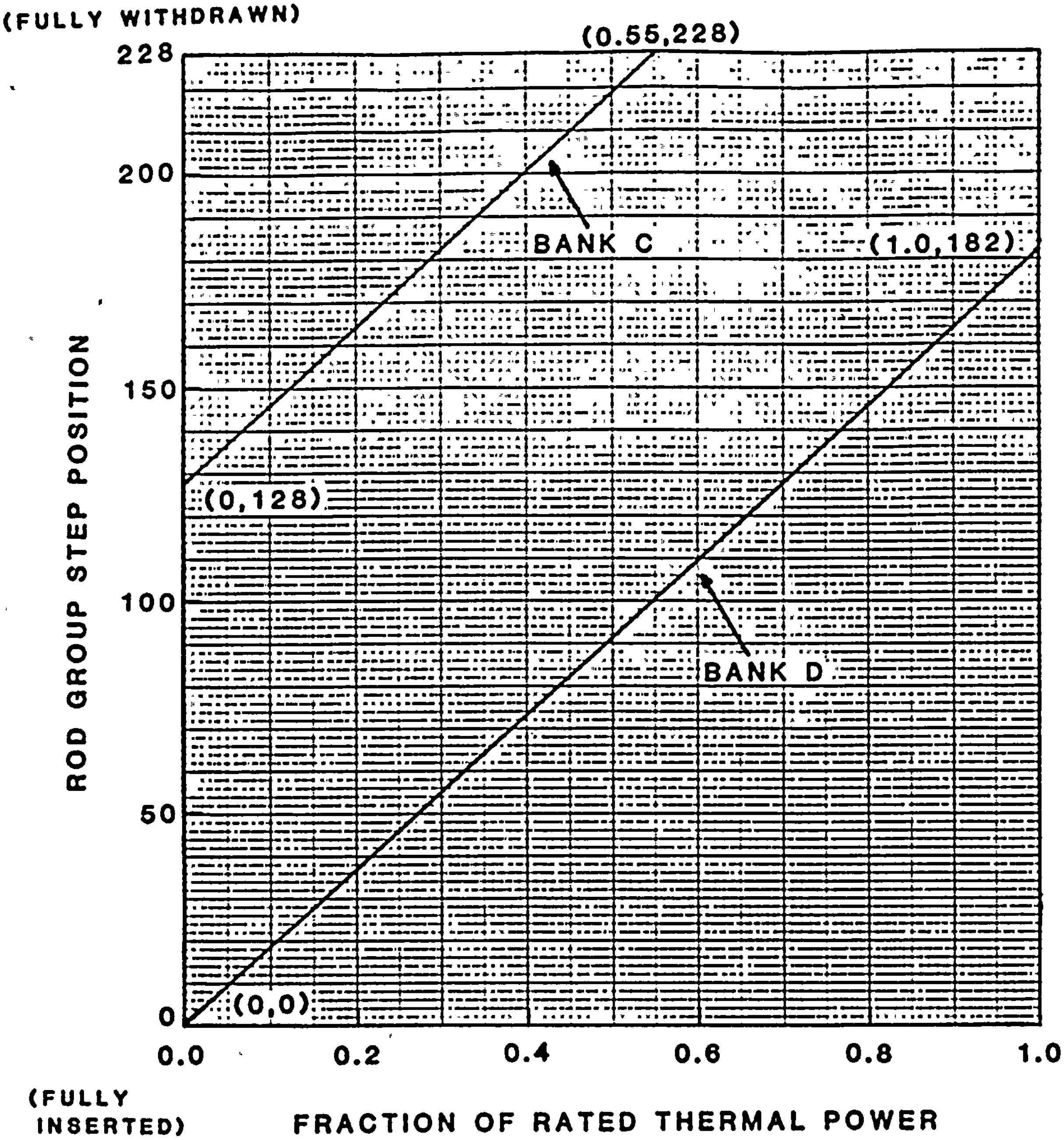


FIGURE 3.1-2 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER 4 LOOP OPERATION

Attachment 9 to AEP:NRC:0916W

EVALUATION OF PEAKING FACTOR LIMIT REPORT TECHNICAL SPECIFICATION PERFORMED BY WESTINGHOUSE ELECTRIC CORPORATION

Westinghouse
Electric Corporation

Nuclear Fuel
Divisions

Box 3912
Pittsburgh Pennsylvania 15230-3912

January 21, 1987
86AE*-G-0008

W-AEP/0322

KEYWORDS:
AEP
PEAKING FACTOR
REPORT
TECH SPEC

Indiana and Michigan Electric Company
c/o Eric G. Lewis
Engineer, Nuclear Materials and Fuel Management
American Electric Power Service Corporation
One Riverside Plaza, 20th Floor
Columbus, OH 43215

Dear Mr. Lewis:

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D. C. COOK UNIT 1
PEAKING FACTOR LIMIT REPORT

American Electric Power (AEP) representatives have asked Westinghouse to support a proposed Tech Spec change to reduce the notification time to the NRC for the Peaking Factor Limit Report (PFLR) from the current 60 days to 15 days. AEP plans to submit the Tech Spec change request by February 1, 1987 so that the change will be in place for the upcoming D. C. Cook Unit 1, Cycle 10.

D. C. Cook 1 has been using FQ Surveillance Tech Specs and has been supplying a V(Z) based PFLR to the NRC for the last cycles. Due to the significantly reduced reload outage time and the cycle design time continually being reduced closer to the operating cycle shutdown, Westinghouse recommends and supports AEP's decision to change their Tech Specs on PFLR notification time from 60 to 15 days. A reduction to 15 days before planned criticality would enable AEP to submit the peaking factor report after the previous cycle shutdown.

Attached is the information requested by AEP.

Very truly yours,

N. E. Campbell

N. E. Campbell
Project Engineer, NFD Projects

NEC:mld

Attachment

cc: M. P. Alexich
J. M. Cleveland
D. H. Malin - w/Enclosure
V. D. Vanderburg

PEAKING FACTOR LIMIT REPORT FOR D. C. COOK UNIT 1, CYCLE 9
F_Q SURVEILLANCE EXAMPLE

This Peaking Factor Limit Report is provided in accordance with Paragraph 6.9.1.11 of the D. C. Cook Unit 1 Technical Specifications.

D. C. Cook Unit 1, Cycle 9 evaluation dependent V(Z) values as a function of burnup are shown in the attached table. This information is sufficient to determine V(Z) versus core height for Cycle 9 burnups in the range of 0 MWD/MTU to 15,750 MWD/MTU through the use of interpolation.

The V(Z) function is used to confirm that the heat flux hot channel factor, $F_Q(Z)$, will be limited to the Technical Specification values of:

$$F_Q(Z) \leq \frac{F_Q^{LIMIT}}{P} [K(Z)] \text{ for } P > 0.50 \text{ and}$$

$$F_Q^{LIMIT} = 2.10$$

$$F_Q(Z) \leq \frac{F_Q^{LIMIT}}{0.50} [K(Z)] \text{ for } P \leq 0.50$$

The appropriate elevation dependent V(Z) values, when applied to a power distribution measured under equilibrium conditions, demonstrates that the initial conditions assumed in the LOCA are met, along with the ECCS acceptance criteria of 10CFR50.46.

(1) WCAP-10216-P-A, Relaxation of Constant Axial Control - F_Q Surveillance Technical Specification

AEP CYCLE 9 - BURNUP DEPENDENT V(Z) FUNCTION

CORE HEIGHT (FEET)	BURNUP (MWD/MTU) 150	1000	2000	4000	6000	8000	10000	12000	14000
0.13	1.07979 *	1.07220 *	1.07625 *	1.08443 *	1.09011 *	1.05319 *	1.09486 *	1.09826 *	1.11886 *
0.38	1.08002 *	1.07737 *	1.08160 *	1.09051 *	1.09563 *	1.06100 *	1.10056 *	1.10399 *	1.12527 *
0.63	1.07827 *	1.07803 *	1.08625 *	1.09345 *	1.09901 *	1.06681 *	1.10338 *	1.10757 *	1.12799 *
0.88	1.08440 *	1.08530 *	1.09036 *	1.09885 *	1.10261 *	1.07377 *	1.10652 *	1.11028 *	1.13074 *
1.13	1.10022 *	1.08967 *	1.09323 *	1.10229 *	1.10739 *	1.08256 *	1.11090 *	1.11557 *	1.13450 *
1.38	1.13063	1.09810	1.09957	1.10243	1.10481	1.08421	1.10707	1.11171	1.12936
1.63	1.13470	1.09424	1.09554	1.09767	1.10094	1.08698	1.10637	1.11232	1.12860
1.88	1.11401	1.09111	1.09049	1.09231	1.09619	1.08652	1.10424	1.10893	1.12431
2.13	1.08956	1.08421	1.08730	1.08867	1.09155	1.08713	1.10056	1.10589	1.11933
2.38	1.07877	1.08046	1.08457	1.08619	1.08772	1.08787	1.09826	1.10231	1.11474
2.63	1.08044	1.07984	1.08303	1.08314	1.08537	1.08818	1.09525	1.09969	1.10890
2.88	1.08132	1.08018	1.08084	1.08021	1.08245	1.08722	1.09174	1.09596	1.10244
3.13	1.07924	1.07847	1.07844	1.07770	1.07914	1.08773	1.08747	1.09176	1.09597
3.38	1.07877	1.07656	1.07708	1.07522	1.07642	1.08819	1.08528	1.08962	1.09602
3.63	1.07850	1.07504	1.07508	1.07394	1.07439	1.09048	1.08454	1.09123	1.09935
3.88	1.07830	1.07519	1.07381	1.07358	1.07518	1.09495	1.08580	1.09288	1.10595
4.14	1.07765	1.07489	1.07376	1.07484	1.07510	1.09823	1.08687	1.09530	1.11256
4.39	1.07722	1.07415	1.07354	1.07490	1.07543	1.10072	1.08758	1.09632	1.11816
4.64	1.07568	1.07443	1.07244	1.07511	1.07562	1.10253	1.08780	1.09714	1.12295
4.89	1.07358	1.07288	1.07185	1.07464	1.07467	1.10389	1.08746	1.09641	1.12610
5.14	1.07304	1.07231	1.07024	1.07344	1.07391	1.10369	1.08650	1.09740	1.12813
5.39	1.07221	1.07135	1.06918	1.07222	1.07198	1.10244	1.08551	1.09943	1.12858
5.64	1.07028	1.06940	1.06725	1.07030	1.07025	1.10007	1.08642	1.10055	1.12663
5.89	1.06740	1.06717	1.06504	1.06825	1.06732	1.09732	1.08666	1.10075	1.12282
6.14	1.06493	1.06443	1.06184	1.06663	1.06706	1.09717	1.08573	1.09877	1.11739
6.39	1.06146	1.06073	1.05841	1.06487	1.06616	1.09505	1.08319	1.09575	1.11020
6.64	1.05723	1.05735	1.05663	1.06297	1.06465	1.09307	1.07986	1.09112	1.10123
6.89	1.05341	1.05565	1.05391	1.05957	1.06122	1.08909	1.07443	1.08490	1.09180
7.14	1.05131	1.05294	1.05226	1.05803	1.05906	1.08307	1.07101	1.07848	1.08908
7.39	1.05023	1.05265	1.05105	1.05640	1.05694	1.07651	1.06662	1.07260	1.08934
7.64	1.04917	1.05009	1.04964	1.05331	1.05489	1.06776	1.06153	1.06590	1.08844
7.89	1.04823	1.04749	1.04678	1.04944	1.05022	1.06245	1.05434	1.05777	1.08569
8.15	1.05234	1.04417	1.04321	1.04497	1.04501	1.05514	1.04710	1.04809	1.08182
8.40	1.05543	1.03884	1.03830	1.03928	1.03877	1.04594	1.03860	1.03772	1.07687
8.65	1.05836	1.03231	1.03288	1.03110	1.03084	1.03707	1.02743	1.02841	1.07163
8.90	1.06137	1.03272	1.02908	1.02758	1.02774	1.03419	1.02824	1.03243	1.07815
9.15	1.06465	1.03439	1.03057	1.03034	1.03104	1.03471	1.03191	1.03460	1.08413
9.40	1.06745	1.03509	1.03230	1.03308	1.03465	1.03530	1.03479	1.03716	1.09017
9.65	1.06490	1.03533	1.03262	1.03530	1.03812	1.03568	1.03670	1.03866	1.09535
9.90	1.06032	1.02941	1.03376	1.03785	1.04153	1.03553	1.03870	1.03985	1.10024
10.15	1.05996	1.02919	1.03096	1.03935	1.04456	1.03392	1.03999	1.04067	1.10413
10.40	1.05964	1.02890	1.02577	1.03866	1.04518	1.03196	1.04054	1.04128	1.10616
10.65	1.06319	1.03192	1.03287	1.04450	1.05115	1.03232	1.04540	1.04528	1.11194
10.90	1.06907 *	1.04168 *	1.04783 *	1.05132 *	1.05735 *	1.03384 *	1.05011 *	1.04781 *	1.11588 *
11.15	1.07523 *	1.04480 *	1.04692 *	1.06071 *	1.07017 *	1.04482 *	1.06676 *	1.06543 *	1.13687 *
11.40	1.07097 *	1.04506 *	1.05253 *	1.07060 *	1.08402 *	1.05712 *	1.08509 *	1.08299 *	1.15243 *
11.65	1.08375 *	1.05781 *	1.06680 *	1.08773 *	1.10107 *	1.07185 *	1.10264 *	1.10213 *	1.17076 *
11.90	1.08491 *	1.05867 *	1.06868 *	1.09092 *	1.10222 *	1.06720 *	1.10076 *	1.09762 *	1.16193 *

* TOP AND BOTTOM 10% EXCLUDED AS PER TECHNICAL SPECIFICATION 4.2.2.2

ATTACHMENT

Attachment 10 to AEP:NRC:0916W

EVALUATION OF PROPOSED MTC LIMIT TECHNICAL SPECIFICATION AND SAFETY LIMIT CURVE FOR 2250 PSIA FOR UNIT 2 SUPPLIED BY EXXON NUCLEAR COMPANY, INC.

SUPPLEMENT TO EVALUATION OF PROPOSED MTC LIMIT TECHNICAL SPECIFICATION

AUG 0 4 1986

EXXON NUCLEAR COMPANY, INC.

600 108TH AVENUE NE, PO BOX 90777, BELLEVUE, WA 98009
(206) 453-4300

July 31, 1986
ENC-AEP/0511

Mr. D. H. Malin, Sr. Engineer
Nuclear Material & Fuel Management
Indiana & Michigan Electric Company
c/o American Electric Power Service Corp.
One Riverside Plaza
Columbus, OH 43216-6631

Subject: Technical Specification Changes to the MTC Limit and Safety Limit Curves

Ref.: (1) Letter, Douglas H. Malin (AEP) to H. G. Shaw (ENC), "D. C. Cook Unit 2, Cycle 6 Required Exxon Fuel Support Activities," dated May 29, 1986 (AEP-ENC/0231)

(2) XN-NF-85-64(P), Revision 1, Supplement 1, "Plant Transient Analysis for D. C. Cook Unit 2 with 10 Percent Steam Generator Tube Plugging," Exxon Nuclear Company, Inc., March 1986.

Dear Doug:

Items 10 and 11 of Reference 1 requested that Exxon Nuclear provide support for Technical Specification (T.S.) changes for D. C. Cook Unit 2 for both the moderator temperature coefficient (MTC) limit and the safety limit curves. The current analyses supporting D. C. Cook Unit 2 Cycle 6 operation, presented in Reference 2, have been reviewed with respect to supporting these changes.

Moderator Temperature Coefficient Limit

The current T.S. gives a MTC limit of +5 pcm/F for all powers less than 70 percent of rated thermal power (RTP) and a limit of 0 pcm/F for all powers of 70 percent or greater. Item 10 of Reference 1 indicates that fuel management flexibility can be gained by replacing the step change in the MTC limit at 70 percent of RTP with a linear ramp rate from +5 pcm/F at 70 percent RTP to 0 pcm/F at 100 percent RTP. Review of the analyses presented in Reference 2 indicates that five transients were performed with a positive MTC at power levels that would potentially be affected by this T.S. change. These five transients are:

15.2.1 Loss of External Load
15.3.1 Loss of Primary Coolant Flow
15.3.3 Locked Primary Pump Rotor
15.4.2 Uncontrolled Rod Withdrawal at Power
15.4.3 Single RCCA Withdrawal

Review of the first three transients (15.2.1, 15.3.1, and 15.3.3) indicated that they had all been performed at 100 percent of RTP with a conservatively high positive MTC value. The review of the 15.4.2 transient analyses showed that the event had been analyzed at three power levels: 9, 60, and 100 percent of RTP. Here, again, the 100 percent RTP case had been performed with a conservatively high positive MTC value consistent with the first four transients. The 9 and 60 percent RTP cases were found to have been performed with temperature-dependent MTC curves, as shown in Reference 2. Both of these cases, however, were adjusted to an initial MTC nominal value of +5 pcm/F for the thermal hydraulic conditions at the start of the transient calculations. These MTC temperature-dependent curves were then conservatively biased for the actual transient calculations.

Review of the fifth transient, 15.4.3, indicated that it had been performed as a bounding analysis of the results obtained in the 15.4.2 transient analyses accounting for the increase in the augmentation factor for a single rod withdrawal. Thus, it supports the same MTC values that are supported by the event 15.4.2.

From the above review, it is apparent that conservatively high positive MTC values have been used in all the transients where it is conservative to do so. Since the positive MTC values used in these analyses either support or exceed the value at the respective power level in the proposed T.S. change, it is concluded that the analyses presented in Reference 2 will support the proposed T.S. change.

Safety Limit Line at 2250 psia

Reference 2 and the current T.S. have safety limit lines (SLL) at pressures of 1840, 1940, 2040, 2290, and 2440 psia. Item 11 of Reference 1 indicates that a SLL is desired at the nominal D. C. Cook Unit 2 operating pressure of 2250 psia. A SLL at 2250 psia has, consequently, been conservatively interpolated from the data that was used in generating the SLLs reported in both Reference 2 and the current T.S. This SLL is shown in the attached figure, and the points defining the SLL are given on the figure.

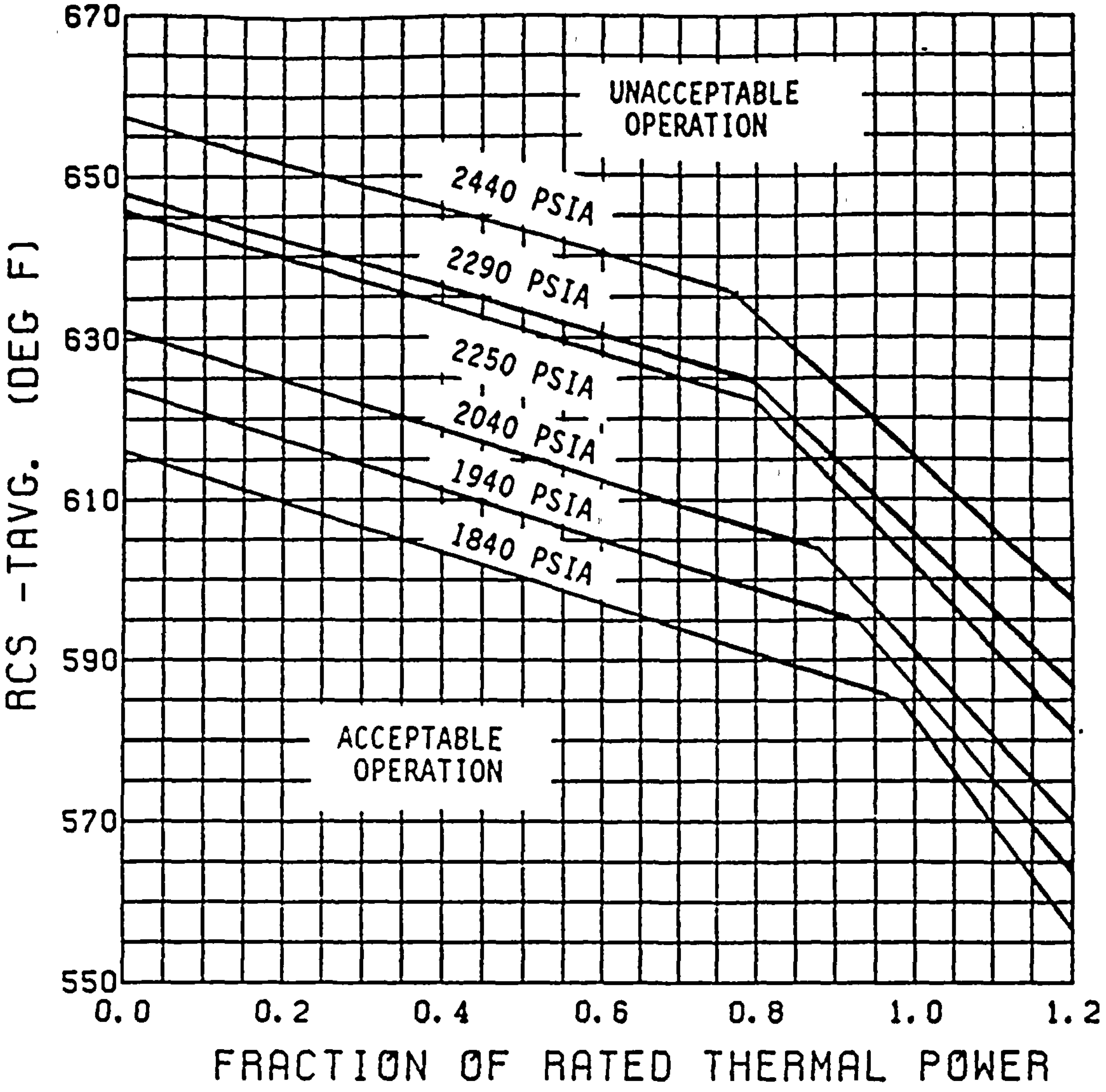
If you have any questions concerning this analysis, please feel free to contact our Jerry Holm at (509) 375-8142.

Sincerely,

H. G. Shaw
Contract Administrator

HGS/wjj

xc: MP Alexich
JM Cleveland
V Vanderburg



PRESSURE (PSIA)	BREAKPOINTS (FRACTION RATED THERMAL POWER, T AVG, DEG F)
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)
2250	(0.00,645.9) , (0.80,622.3) , (1.20,580.9)
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)

MAR 1 1 1987

ADVANCED NUCLEAR FUELS CORPORATION

600 108th AVENUE NE, PO BOX 90777, BELLEVUE, WA 98009-0777
(206) 453-4300

March 5, 1987
ANF-AEP/0557

Mr. Richard B. Bennett, Engineer
Nuclear Materials & Fuel Management
Indiana & Michigan Electric Company
c/o American Electric Power Service Corp.
One Riverside Plaza, 20th Floor
Columbus, OH 43216-6631

Ref.: (1) Letter, H.G. Shaw (ANF) to D.H. Malin (AEP), "Technical Specification Changes to the MTC Limit and Safety Limit Curves," dated July 31, 1986 (ENC-AEP/0511)

(2) XN-NF-85-64, Rev. 1, "Plant Transient Analysis for D.C. Cook Unit 2 with 10% Steam Generator Tube Plugging," Exxon Nuclear Company, March 1986

(3) XN-NF-85-64, Rev. 2, Supp. 1, "Plant Transient Analysis for D.C. Cook Unit 2 with 10% Steam Generator Tube Plugging," Exxon Nuclear Company, September 1986

(4) Letter, G.N. Ward (ANF) to H.R. Denton (NRC), "Response to NRC Questions on XN-NF-85-28(P)," dated April 14, 1986 (GNW:053:86)

Dear Mr. Bennett:

This letter is in response to your request in a telephone conversation with Jerry Holm on February 26, 1987 for an additional evaluation of the proposed Technical Specification (T.S.) change in the D.C. Cook Unit 2 moderator temperature coefficient (MTC). Specifically, an evaluation of the T.S. change on events

15.1.1	Decrease in Feedwater Temperature
15.1.2	Increase in Feedwater Flow
15.1.5	Steam Line Break
15.4.1	RCCA Withdrawal from Subcritical
15.4.6	Boron Dilution
15.4.8	RCCA Ejection

was requested. The initial evaluation of the proposed T.S. change was reported in Reference 1 and the evaluation of these additional events is presented in the following paragraphs.

The proposed T.S. change in the MTC involves the replacement of a step change in MTC at 70% rated thermal power (RTP) from +5 pcm/°F to ≤0 pcm/°F for all

RTP greater than 70% to a ramp change from +5 pcm/°F at 70% RTP to ≤0 pcm/°F at 100% RTP. This proposed T.S. change will allow a positive MTC over the power range from 70% to 100% of RTP, whereas only a 0 or negative MTC was allowed before. A positive MTC is only a concern for heatup events since for these events the potential for an increase in power is aggravated by a positive reactivity contribution from the MTC. Events 15.1.1, 15.1.2 and 15.1.5 are all cooldown events, and are consequently limiting only for negative MTCs. Therefore, these three events are unaffected by the T.S. change and will continue to be bounded by the current analysis presented in Reference 2.

The event 15.4.1, RCCA Withdrawal from Subcritical or Low Power, is not affected by the proposed T.S. change. The limiting case is initiated from a low initial power level (approximately 1.0E-9 RTP), which bounds the hot shutdown and startup modes of operation. This low initial power level yields the maximum margin to trip, and hence the maximum time for rod withdrawal. These two conditions produce the largest prompt multiplication which maximizes the power overshoot past trip. Since the proposed T.S. change only affects operation at or above 70% RTP, the limiting event presented in Reference 2 remains bounding.

The Boron Dilution event (15.4.6) was evaluated for the full range of operating modes, that is, for all modes from 1 to 6. Modes 2 through 6 are all restricted to power levels less than 5% of RTP, and are consequently unaffected by the proposed T.S. change. Mode 1, which is power operation with powers greater than 5% of RTP, is bounded by Event 15.4.2, RCCA Withdrawal at Power, which was addressed in Reference 1. It is bounded by Event 15.4.2 from a DNB standpoint because the reactivity insertion rates considered in 15.4.2 bound the maximum rate achievable by boron dilution. The time to lose shutdown margin in Mode 1 is unaffected by the T.S. change since it is only a function of the shutdown margin, primary coolant system volume, and the maximum boron dilution rate. Since none of these parameters are altered by the T.S. change, the analysis presented in Reference 3 remains bounding.

The limiting RCCS Ejection event (15.4.8) was found to occur at end of cycle (EOC) from HFP conditions. The EOC conditions were found to be limiting over the BOC conditions due to a larger rod worth and a smaller delayed neutron fraction at EOC. Both these conditions result in an increase in the calculated return to power for the event. The proposed T.S. change will not affect the results of the EOC analysis from HFP conditions because the MTC is negative at EOC. Furthermore, the MTC has only a small effect on the results of this event because the extremely rapid nature of the event does not allow sufficient time for the heat to be transferred from the fuel. Thus, the current analysis for this event presented in Reference 4 will not be altered by the proposed T.S. change and will continue to bound current operating conditions.

If you have any further questions regarding this MTC Technical Specification review, please feel free to contact our Mr. Jerry Holm (509-375-8142).

Sincerely,

J S Holm for

H. G. Shaw
Contract Administrator

gf

cc: Mr. J.M. Cleveland
Mr. D.H. Malin
Mr. V. VanderBurg
Mr. J.S. Holm (ANF)

Attachment 11 to AEP:NRC:0916W

EVALUATION OF THE IMPACT OF 2000 GPM PRIMARY FLOW ON THE UNIT 2

DILUTION TRANSIENT PERFORMED BY EXXON NUCLEAR COMPANY, INC.

JUL 17 1986

EXXON NUCLEAR COMPANY, INC.

600 108TH AVENUE NE, PO BOX 90777, BELLEVUE, WA 98009
(206) 453-4300

July 11, 1986
ENC-AEP/0505

Mr. D. H. Malin, Sr. Engineer
Nuclear Material & Fuel Management
Indiana & Michigan Electric Company
c/o American Electric Power Service Corp.
One Riverside Plaza
Columbus, OH 43216-6631

Subject: Boron Dilution Analysis During RHR Operation for D.C. Cook Unit 2

Ref.: (1) Letter, D. H. Malin (AEP) to H. G. Shaw (AEP), "D.C. Cook Unit 2, Cycle 6 Required Exxon Fuel Support Activities," dated May 29, 1986 (AEP-ENC/0231)

(2) XN-NF-85-64(P), Rev. 1, Supp. 1, "Plant Transient Analysis for D.C. Cook Unit 2 with 10% Steam Generator Tube Plugging," Exxon Nuclear Company, Inc., March 1986

Dear Doug:

Item 6 of Reference 1 requested that Exxon Nuclear perform a boron dilution analysis to support operation of D.C. Cook Unit 2 with a residual heat removal (RHR) system flow rate of 2000 gpm. The analyses presented in Reference 2 were performed to support an RHR flow rate of 3000 gpm, which is the minimum flow rate specified in the D.C. Cook Unit 2 Technical Specifications.

The RHR analyses described in Reference 2 were performed using a dilution front method since the RHR flow rate is potentially insufficient to assure a completely mixed primary coolant volume. This dilution front method assumes a step boron concentration reduction at the charging inlet which migrates through the core and the remainder of the non-stagnant primary coolant and RHR system. When this dilution front completes one transit time, the entire volume of the non-stagnant coolant system is at the reduced boron concentration and a second step reduction begins to transit the system.

A detailed review of the calculations which have been performed indicates that the analysis presented in Reference 2 will bound RHR flow rates 2000 gpm or greater. The RHR flow rate is not specified in Reference 2. A revision to this report will be issued which specifies the minimum flow rate.

If you have any questions concerning this analysis, please feel free to contact our Mr. Jerry Holm, telephone 509-375-8142.

Sincerely,

J S Holm for

H. G. Shaw
Contract Administrator

gf

cc: Mr. M. P. Alexich
Mr. J. M. Cleveland
Mr. V. Vanderburg

Attachment 12 to AEP:NRC:0916W

PRESSURIZER PRESSURE READABILITY ALLOWANCE

AND RCS FLOW MEASUREMENT ALLOWANCE FOR UNIT 2

PREVIOUSLY SUBMITTED WITH AEP:NRC:0916I

Attachment 7 is provided as an aid to assist the reviewer in understanding the development of certain values cited in the Technical Specifications. The included calculations supplement information provided in XN-NF-85-64(P), XN-NF-85-64(P) Rev. 1, and WCAP 11080. Reference to Attachment 7 is indicated in the Remarks column of Attachment 10 for those Technical Specification items which require the additional explanation so provided.

Item A of this attachment demonstrates the development of the Reactor Coolant System (RCS) Analysis Flow Value.

Item B of this attachment demonstrates the derivation of the required minimum indicated RCS Flow in lbm/hr.

Item C of this attachment demonstrates the conversion of the minimum indicated RCS flow obtained in Item B from lbm/hr to gpm.

Item D1 provides the minimum indicated pressurizer pressure indication value in psig for Mode 1 operation.

Item D2 provides the minimum indicated pressurizer pressure indication value in psig for Modes 2 & 3 operation.

ATTACHMENT 7

A. ANALYSIS VALUE OF REACTOR COOLANT SYSTEM (RCS) FLOW

Nominal RCS Flow with 10% Steam Generator Tube Plugging	141.3×10^6 lbm/hr
Flow Measurement Uncertainty (3.5%)	5.0×10^6 lbm/hr
Flow Measurement Repeatability	3.4×10^6 lbm/hr
Analysis Flow: 141.3 E6 - 5.0 E6 - 3.4 E6 =	132.9×10^6 lbm/hr

B. TECHNICAL SPECIFICATION MINIMUM INDICATED REACTOR COOLANT SYSTEM (RCS) FLOW (lbm/hr)

Nominal RCS Flow with 10% Steam Generator Tube Plugging	141.3×10^6 lbm/hr
Flow Measurement Repeatability	3.4×10^6 lbm/hr
Correction to Flow Measurement Repeatability to Support Larger Pressure Allowance (Section 15.0.2, XN-NF-85-64(P) Rev. 1)	0.7×10^6 lbm/hr
Modified Flow Measurement Repeatability: 3.4 E6 - 0.7 E6 =	2.7×10^6 lbm/hr
Minimum Indicated RCS Flow: 141.3 E6 - 2.7 E6 =	138.6×10^6 lbm/hr

C. TECHNICAL SPECIFICATION MINIMUM INDICATED REACTOR COOLANT SYSTEM (RCS) FLOW (gpm)

Minimum Indicated RCS Flow	138.6×10^6 lbm/hr
RCS Pressure	2250 psia
T_{cold}	542.3°F
1 Gallon =	0.13368 cu.ft.
Specific Volume of Water at Stated Pressure and Temperataure Conditions (1967 ASME Steam Tables)	0.021119 ft^3/lbm

RCS Flow = (138.6 E6 lbm/hr) X (1 hr/60 min) X (0.021119 ft^3/lbm) X (1 gal/0.13368 ft.)

Minimum Indicated RCS Flow =	364,940 gpm
Minimum Indicated RCS Flow/Loop =	91,240 gpm

D. INDICATED PRESSURIZER PRESSURE DNB LIMIT (TABLES 3.2-1 AND 3.2-2)

The method of determining the allowance for pressure readability is similar to that provided in WCAP 11080 for the indicated T_{avg}. Actual values for the terms used in the calculation, with the exception of the rack calibration allowance and the indicator readability, were also obtained from WCAP 11080 Page viii. The value used for the rack calibration allowance was obtained from the pressurizer pressure channel calibration procedure; the value used for indicator readability was determined from a review of the indicator span and scale.

The total pressurizer pressure channel allowance was determined to be 3.41% of span which equates to 27.29 psia.

Assuming a minimum of 3 channels available for averaging, the allowance may be reduced by the square root of 3. This yields a final pressurizer pressure readability allowance of 15.8 psia.

1) Minimum Indicated Pressure in Mode 1

Nominal Pressure =	2250 psia
Pressure Control Allowance (WCAP 11080, Page 3) =	Proprietary
Indication Allowance =	15.8 psi
Allowance assumed in Analysis =	40 psi
Additional Pressure Allowance accounted for by .5% increase in minimum RCS Flow (Section 15.0.2 XN-NF-85-64(P) Rev. 1) =	7.5 psi
Analysis Pressure: 2250 - 40 - 7.5 =	2202.5 psia
Minimum Indicated Pressurizer Pressure: 2202.5 + 15.8 =	2218.3 psia
Table 3.2-1 Value for Minimum Indicated Pressure in Mode 1: 2220 psia =	2205 psig

2) Minimum Indicated Pressure in Modes 2 & 3

Analysis Pressure	2175 psia
Minimum Indicated Pressure: 2175 + 15.8 =	2190.8 psia
Table 3.2-2 Value for Minimum Indicated Pressure in Modes 2 & 3: 2191 psia =	2176 psig

Attachment 13 to AEP:NRC:0916W

SUMMARY TO ATTACHMENT 13

ANALYSIS TO JUSTIFY AN INCREASE IN
BORON CONCENTRATION IN THE REFUELING WATER STORAGE TANKS AND
ACCUMULATOR TANKS PERFORMED BY WESTINGHOUSE ELECTRIC CORPORATION

EVALUATIONS OF THE IMPACT ON AN INCREASE IN BORON CONCENTRATION ON
THE UNIT 2 ANALYSES PERFORMED BY EXXON NUCLEAR CORPORATION
AND BY AMERICAN ELECTRIC POWER SERVICE CORPORATION

SUMMARY OF ATTACHMENT 13

The purpose of Attachment 13 is to provide justification for increasing the minimum boron concentration in the RWST and accumulators for D. C. Cook Units 1 and 2. The minimum boron concentration is being increased to 2400 ppm to provide fuel management flexibility. The changes impact T/Ss 3.1.2.7 (Borated Water Sources - Shutdown), 3.1.2.8 (Borated Water Sources - Operating), 3.5.1 (Accumulators), and 3.5.5 (Refueling Water Storage Tank).

Included in this attachment is a safety evaluation performed by Westinghouse in support of this change. The Westinghouse evaluation considers the impact that raising the minimum boron concentration has on the LOCA and non-LOCA safety analyses, as well as LOCA related design considerations. The Westinghouse discussion of LOCA related design considerations references WCAP 11020, entitled "Spray Additive Tank Deletion Analysis for the Donald C. Cook Nuclear Plant". This analysis was submitted to the NRC via our letter AEP:NRC:0914C, dated February 28, 1986, in support of our proposal to remove the NaOH spray additive tank and its associated T/S (T/S 3/4.6.2.2). Although this submittal is still under NRC review, reference was made to it since operation without spray additive is bounding with respect to those issues considered in the LOCA related design considerations section of the Westinghouse evaluation contained in this attachment.

The Westinghouse evaluation also contains a discussion of post-LOCA long term core cooling. This discussion demonstrates that for D. C. Cook Unit 1 during the present and upcoming fuel cycles, the boron concentration in the sump following a LOCA would be sufficient to maintain the reactor subcritical.

Analogous evaluations for the present Unit 2 fuel cycle were performed by the American Electric Power Service Corporation, using methodology similar to that described by Westinghouse. These calculations are described in our letter AEP:NRC:1008, which was submitted to the NRC on November 17, 1986.

Related to the change in RWST and accumulator boron concentrations are changes to the boric acid storage tank and RWST volumes required by T/Ss 3.1.2.7 and 3.1.2.8. These changes are described by Westinghouse in Attachment 13. The changes to the required tank volumes ensure the capability to bring the core from hot, full power to Mode 4 and 6 shutdown conditions, including allowing for the increased shutdown margin requirements based on the boron dilution event. (Reference our proposed Unit 1 T/Ss 3/4.1.1.2). Similar changes were made for Unit 2 and approved in Amendment 82 to DPR-74.

Lastly, Attachment 13 contains a letter from Advanced Nuclear Fuels Corporation (ANF, formerly Exxon Nuclear Co.). This letter documents ANF's concurrence with the increase in the RWST and accumulator boron concentration.

SAFETY EVALUATION FOR

INCREASE IN THE BORON CONCENTRATION LIMITS

FOR THE RWST AND ACCUMULATOR LIMITS

FOR

D. C. COOK UNITS 1 AND 2

1.0 INTRODUCTION

It must be demonstrated, each cycle, that the core can be maintained subcritical via boron addition from the ECCS in the unlikely event of a Large Break LOCA. However, evaluations of future fuel cycles show that subcriticality may not be assured with the present minimum RWST/Accumulator boron concentration. In order to provide adequate post-LOCA shutdown margin for future cycles, increasing the accumulator and RWST boron concentration into the range of 2600 ppm is proposed.

2.0 SCOPE OF EVALUATION

Both Westinghouse Electric Corporation and American Electric Power Service Corporation (AEPSC) have assessed the impact of increasing the RWST and accumulator boron concentration from a minimum of 1950 ppm into the range of 2600 ppm. This assessment identified the following areas in which the boron concentration increase must be shown to have a favorable or non-detrimental impact on the D. C. Cook design basis:

1. Non-LOCA Safety Analysis

2. LOCA Analysis (10 CFR50.46)
    - Small Breaks
    - Large Breaks
    - Long-Term Core Cooling
    - Boron Precipitation

3. LOCA Related Design Consideration
   - Radiological Consequences
   - Hydrogen Production
   - Equipment Qualifications

Evaluation summaries for each of the above areas are provided in the following section.

3.0 SAFETY EVALUATION

3.1 FSAR NON-LOCA SAFETY ANALYSIS

The proposed increase in RWST boron concentration has been evaluated and the impact of this change on each of the non-LOCA FSAR transients which model the RWST and/or accumulators follows.

3.1.1 Uncontrolled Boron Dilution

The refueling and startup cases are impacted by the RWST boron concentration change. The increased concentration increases the time to reach criticality which increases the available operator action time. This is a benefit in the analysis.

3.1.2 Major Secondary System Pipe Rupture

a. Rupture of a Main Steamline Core Response and Mass/Energy Release Inside Containment - The current safety analyses for Units 1 and 2 assumes that boron concentration of 20,000 ppm in the Boron Injection Tank (BIT) would be available to provide negative reactivity to shut down the reactor. Although an increase in boron concentration in the RWST and accumulators would generally be a benefit for this transient, the impact would be negligible when compared with the available BIT boron concentration (20,000 ppm) which would be purged before the RWST water reaches the core. As such, the current safety analyses provided in Chapter 14 of the FSAR for the core response and the mass and energy release inside containment remain valid.

b. Rupture of a Main Steamline Mass/Energy Release Outside Containment - The recent outside containment mass/energy release data following a steamline break provided in WCAP-10961 Revision 1 (Steamline Break Mass/Energy Releases for Equipment Qualification Outside Containment) assumed a BIT boron concentration of 0 ppm to bound the other similar Westinghouse units. An increase in the minimum boron concentration in the RWST and accumulators would be a benefit for this transient because it would shut down the reactor sooner. The boron concentration increases would give less limiting results for the mass/energy releases outside containment provided in WCAP-10961.

3.1.3 Accidental Depressurization of the Main Steam System

The current safety analyses for Units 1 and 2 assumes a boron concentration of 20,000 ppm in the Boron Injection Tank (BIT) would be available to provide negative reactivity to shut down the reactor. Although an increase in boron concentration in the RWST and accumulators would generally be a benefit for this transient, the impact would be negligible when compared with the available BIT boron concentration (20,000 ppm) which would be purged before the RWST water reaches the core. As such, the current safety analysis provided in Chapter 14 of the FSAR remain valid.

3.1.4 Conclusions

The above discussions demonstrate that the proposed RWST and accumulators boron concentration increase does not adversely impact the conclusions of non-LOCA transient analyses. Accident reanalysis is not required, therefore there are no FSAR changes associated with this evaluation.

3.2 FSAR LOCA ANALYSIS

The following evaluation discusses the impact of the increase from 1950 ppm to 2400 ppm in RWST/Accumulator boron concentrations for D. C. Cook Units 1 and 2 on the Large and Small break LOCA analyses, Long Term Core Cooling and Hot Leg Switchover Time. The time when hot leg recirculation should be initiated to

prevent boron precipitation in the core was determined to be 12 hours following a LOCA. FSAR section 6.3 was revised to reflect the hot leg switchover time for both units.

3.2.1 Hot Leg Recirculation Switchover Time

An analysis has been performed to determine the maximum boron concentration in the reactor vessel following a hypothetical LOCA. This analysis considered D. C. Cook Units 1 and 2 with a proposed maximum boric acid concentration of 2600 ppm in the RWST, accumulators, and RCS.

The analysis considers the increase in boric acid concentration in the reactor vessel during the long term cooling phase of a LOCA, assuming a conservatively small effective vessel volume. This volume includes only the free volumes of the reactor core and upper plenum below the bottom of the hot leg nozzles. This assumption conservatively neglects the mixing of boric acid solution with directly connected volumes, such as the reactor vessel lower plenum. The calculation of boric acid concentration in the reactor vessel considers a cold leg break of the reactor coolant system in which steam is generated in the core from decay heat while the boron associated with the boric acid solution is completely separated from the steam and remains in the effective vessel volume.

The results of the analysis show that the maximum allowable boric acid concentration of 23.53 weight percent established by the NRC, which is the boric acid solubility limit less 4 weight percent, will not be exceeded in the vessel if hot leg injection is initiated 12 hours after the LOCA inception. This switchover time is applicable to both units. The operator should reference this switchover time against the reactor trip/SI actuation signal. The typical time interval between the accident inception and the reactor trip/SI actuation signal is negligible when compared to the switchover time.

Procedures philosophy assumes that it would be very difficult for the operator to differentiate between break sizes and locations. Therefore one hot leg switchover time is used to cover the complete break spectrum.

3.2.2 Small Break LOCA D. C. Cook Unit 2

The current FSAR small break analysis for D. C. Cook Unit No. 2 employs the Westinghouse WFLASH Evaluation Model and is based on a full core of Westinghouse fuel. Since the time that the FSAR small break LOCA analysis for D. C. Cook Unit No. 2 was performed, the Westinghouse fuel has been almost completely replaced with fuel provided by the Exxon Nuclear Corporation (ENC). The Peak Clad Temperature results of small break LOCA analyses employing this Evaluation Model will not be altered by the changes in boron concentrations for the RWST and accumulators. Confirmation of the applicability of the FSAR small break LOCA analysis will be required by the current fuel vendor.

3.2.3 Small Break LOCA D. C. Cook Unit 1

Small break LOCA analyses performed by Westinghouse assume that the reactor core is brought to a subcritical condition by the trip reactivity of the control rods. There is no assumption requiring the presence of boron in the ECCS water or needing the negative reactivity provided by the soluble boron. Thus the changes to the RWST and Accumulator Tech-Specs covering boron concentrations do not alter the conclusions of the FSAR small break LOCA analysis.

3.2.4 Large Break LOCA D. C. Cook Unit No. 1

Large break LOCA analyses performed by Westinghouse do not take credit for the negative reactivity introduced by the soluble boron in the ECCS water in determining reactor power level during the early phases of the hypothetical large break LOCA. The large break LOCA analyses performed by Westinghouse analyze the LOCA transient to a time just beyond the time at which Peak Cladding Temperature is calculated to occur. During this time period the reactor is kept subcritical by the voids present in the core. Thus the changes to the RWST and Accumulator Tech-Specs covering boron concentrations do not alter the conclusions of the FSAR large break LOCA analyses.

3.2.5 Large Break LOCA D. C. Cook Unit No. 2

It is the responsibility of the current fuel vendor to address the impacts that the proposed Tech-Spec changes may have on the fuel, LOCA model, LOCA methodology, and LOCA assumptions employed for this unit.

3.2.6 Long Term Core Cooling D. C. Cook Unit No. 1

The Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long-term cooling" is defined in WCAP-8339 (page 4-22). The Westinghouse commitment is that the reactor remain shutdown by the borated ECCS water. Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the RWST and Accumulators must have a concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical assuming all control rods out (ARO). The attached figure (Figure 1) shows the effect on the post-LOCA RCS/Sump boron concentration as a result of changing the minimum Tech-Spec boron concentration from 1950 to 2400 for the RWST and from 1950 to 2400 for the Accumulators. The result is an increase of over 200 PPM in the RCS/Sump boron concentration which would provide enough negative reactivity to keep the cycle 9 core subcritical with a margin of about 204 PPM. Thus the long-term core cooling requirement that the reactor remain subcritical is satisfied by the new proposed Technical Specifications for D. C. Cook Unit No. 1. It is here noted that the ability to maintain core subcriticality following a hypothesized LOCA is highly dependent on cycle specific core conditions, and an evaluation of Long Term Core Cooling capability is routinely performed before the start-up of each cycle.

3.2.7 Long Term Core Cooling D. C. Cook Unit No. 2

The Westinghouse licensing position for satisfying the requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) "Long-term cooling" is defined in WCAP-8339 (page 4-22). The assumptions employed by Westinghouse to satisfy these requirements have been stated above (LONG TERM CORE COOLING D. C. COOK UNIT NO. 1). The assumptions employed by ENC for the satisfaction of the

requirements of 10CFR Part 50 Section 50.46 Paragraph (b) Item (5) may differ from those employed by Westinghouse. At the request of American Electric Power, Westinghouse has performed a calculation to determine the minimum Post-LOCA RCS/Sump boron concentration for a range of pre-trip RCS boron concentrations for D. C. Cook Unit No. 2. This calculation is based on the current Westinghouse assumptions and methodology for Westinghouse fuel using the most recent available input sources for D. C. Cook Unit No. 2. The attached figure (Figure 2) shows the minimum post-LOCA RCS/Sump boron concentration as a function of pre-trip RCS boron concentration with a minimum Tech-Spec boron concentration of 2400 PPM for the RWST and 2400 PPM for the Accumulators based on the above-stated assumptions. The adequacy of these limits to ensure core subcriticality following a postulated large break LOCA is dependent on the limiting RCS boron requirements for criticality ad dictated by the core design for a specific cycle. Confirmation of the applicability of these limits and that Long Term Core Cooling requirements will be satisfied must be provided by American Electric Power.

3.2.8 Conclusions

The increase in the minimum RWST boron concentration from 1950 PPM to 2400 PPM and minimum Accumulator boron concentrations from 1950 to 2400 do not negatively affect the FSAR LOCA analysis for D. C. Cook Unit No. 1. The new proposed Technical Specifications provide an additional safety margin to ensure long-term cooling of the reactor core after a postulated large break LOCA for D. C. Cook Unit No. 1.

3.3 LONG TERM SUMP pH

The minimum calculated pH is 7.6. The assumptions used in calculating this sump pH are as follows:

1. The amount of boric acid that is transported to the sump was maximized. The volumes of solution that were assumed to enter the sump are as follows:

    a. RWST total tank volume as provided in chapter 6 of the FSAR,

b. maximum SI accumulator water volume as allowed by technical specifications,

c. boron injection tank volume of 900 gallons,

d. RCS volumes and auxiliary piping volumes as indicated by Westinghouse calculations.

The boric acid concentration of solutions entering the sump was maximized. The following concentrations were assumed:

a. The maximum allowable RWST concentration was assumed to be 2600 ppm.

b. The accumulator and piping concentration was assumed equal to the RWST concentration.

c. The maximum boron injection tank concentration allowed by technical specifications was used.

d. The RCS concentration was conservatively chosen as 2400 ppm.

2. The amount of sodium tetraborate transported to the sump was minimized by assuming the minimum ice mass and ice pH allowed by the technical specifications.

These assumptions taken in total were aimed at determining a conservative lower bound for the long term sump pH.

3.4 RADIOLOGICAL, HYDROGEN, AND EQUIPMENT QUALIFICATION EVALUATIONS

Increasing the boron concentration in the Refueling Water Storage Tank (RWST) and accumulators decreases the pH of the recirculating core cooling solution. A decrease in pH can decrease the elemental iodine decontamination factor (DF), increase the rate of hydrogen production due to corrosion of zinc (galvanize and zinc based paint) and increase the potential for chloride induced stress corrosion cracking of stainless steel.

Based on the above considerations, 2600 ppm has been determined to be the maximum RWST and accumulator boron concentration. Details of the specific evaluations follow.

3.5 RADIOLOGICAL CONSEQUENCES

The minimum calculated sump pH of 7.6 is sufficient to support the elemental iodine DF assumed in the Spray Additive Deletion Analysis (reference 1). Hence, the radiological consequences will not change as a result of the boron increase, and the dose analysis (reference 1) remains valid.

The reference analysis assumes a DF of 1000 (99.9 percent removal) for the combined elemental iodine reduction effects of the ice condenser, sprays, surface deposition, and radioactive decay. The sump solution, with a minimum pH of 7.6, can retain approximately 98 percent (reference 2) of the elemental iodine that is assumed to be released from the core. The containment surfaces utilized for deposition have the capacity to retain 100 percent of the released iodine in the short term and greater than 70 percent in the long term. Hence, the DF assumption of the reference analysis, for the combined long-term iodine capacity of sump and surfaces, remains valid.

3.5.1 Sump pH

The calculation of the minimum equilibrium sump solution pH considers the following delivered tank volumes, ice mass, and boron concentrations:

RWST - 420,000 gal, 2600 ppm B
Accumulators(4) - 29,052 gal, 2600 ppm B
RCS (hot zero power, no xenon) - 88, 958 gal, 2400 ppm B
Sodium tetraborate ice - 2,372,000 lb, 1800 ppm B
Boron injection tank - 900 gal, 22,500 ppm B

The resulting pH is 7.6, which is sufficient to support a partition coefficient of approximately 600 (reference 2) which supports an elemental iodine DF of 78 (98% capacity) for the recirculating solution.

3.6 HYDROGEN PRODUCTION

Hydrogen produced by the corrosion of aluminum and zinc is a function of solution pH. The corrosion rates incorporated in the FSAR Chapter 14 combustible gas analysis were assumed to be based on a spray pH of 9.3 and 2000 ppm B.

The evaluation of hydrogen production presented in reference 1 concludes that aluminum corrosion decreases with decreasing pH and zinc corrosion increases. Specifically, the zinc corrosion rate at pH 5 was found to be as much as 20 percent greater than the pH 9.3 rate for the temperature range of 110 to 175 degrees F (see attached Figure 6-1 of the referenced report). However, it was further concluded that this low temperature increase would have a negligible impact on the aggregate hydrogen generation rate since the solution pH would be quickly raised into the caustic range by the melting sodium tetraborate ice.

Additionally, a corrosion rate constant comparison was made for the FSAR condition (pH 9.3, 2000 ppm B) versus the new reduced pH/increased boron condition (pH 7.6, 2400 to 2600 ppm B) (reference 3). The comparison showed a rate constant change, for the new condition, of +1 to - 0.5 percent, depending on temperature. This variation is also concluded to have a negligible impact on the aggregate hydrogen generation rate.

To summarize, the rates of hydrogen generation due to corrosion of aluminum and zinc, for the increased boron/decreased pH condition, are enveloped by the analysis presented in the FSAR.

3.7 EQUIPMENT QUALIFICATION

The primary concerns of equipment qualification are protection of the stainless steel components of the emergency core cooling system from chloride induced stress corrosion cracking, failures of electrical components required to operate post-accident, and failures of containment coatings which could jeopardize the ECCS by flaking or pealing off, clogging the emergency sump and other flow paths, and thus restrict the flow of emergency core cooling water.

3.8 PROTECTION OF STAINLESS STEEL

To minimize the occurrence of chloride stress corrosion cracking of stainless steel, Westinghouse recommends maintaining the equilibrium sump solution pH above 7.5 (Reference 4). The NRC recommends a solution pH in the range of 7 to 9.5 (Reference 5). The minimum calculated sump solution pH of 7.6 is consistent with these recommendations.

3.9 ELECTRICAL COMPONENTS

Electrical equipment is tested to determine the ability of component seals to exclude the containment environment from the interior of the component. To maximize the challenge to the seal materials, high pH sprays, in the range of 8 to 11, have traditionally been used.

For all modes of ECCS operation, the solution pH with increased boron concentration will always be less than the corresponding pH with reduced boron. Hence, components qualified at higher pH may actually have a longer post-accident service life in a lower pH (in the caustic range) environment.

3.10 CONTAINMENT COATINGS

Coatings are used in the containment to provide corrosion protection for metals and to aid in decontamination of surfaces during normal operation.

Like electrical equipment, coatings are tested with a high pH solution to maximize the potential deterioration of the coating, and may show better resistance to lower pH solutions.

4.0 IMPACT ON D. C. COOK (UNITS 1 & 2) TECHNICAL SPECIFICATION

The D. C. Cook Technical Specification that were affected by increasing the RWST and accumulator allowable boron concentrations are presented here via marked up technical specification pages.

5.0 FINAL SAFETY ANALYSIS REPORT (FSAR)/TECHNICAL SPECIFICATION CHANGES

Please find attached the FSAR/Technical Specification changes that were modified as a result of the RWST and accumulator boron concentration increase. The ph limits in the basis of Specification 3.5.5 were also revised to reflect the increase in the boron concentration.

Changes to the Boration Systems basis (3/4.1.2) resulted from recalculating RWST volumes based on a boron concentration of 2400 ppm and bounding boron requirements for D. C. Cook Unit 1 extended fuel cycles. These changes include the additional borated water source volumes required to consider a boron dilution event during cooldown from HFP to 200 degrees-F (Mode 4) and cooldown from 200 degrees-F to 140 degrees-F (Mode 6).

Changes to the Boric Acid Tank (BAT) and RWST volumes in Specifications 3.1.2.7 and 3.1.2.8 and in the Boration Systems basis (3/4.1.2) are associate with the above boron dilution event.

6.0 SUMMARY AND CONCLUSIONS

The proposed increase in the RWST and accumulator allowable boron concentration limits to 2600 ppm has been assessed from a safety standpoint. Based on these results, it is concluded the proposed boron concentration increases will have no adverse impact on the non-LOCA Accident Analysis, the LOCA Analysis or LOCA Related Design Considerations and is thus acceptable for implementation at D. C. Cook.

Confirmation that the boron concentration increases will provide enough margin to meet post-LOCA shutdown requirements will be concluded through the normal Westinghouse RSAC evaluation process.

7.0 REFERENCES

1. "Spray Additive Tank Deletion Analysis for the Donald C. Cook Nuclear Plant", WCAP-11020 (WCAP-11021, non-proprietary), December, 1985.

2. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", NUREG-0800, Section 6.5.2.

3. "The Relative Importance of Temperature, pH and Boric Acid Concentration on Rates of H_2 Production From Galvanized Steel Corrosion", NUREG/CR-2812, November, 1983.

4. "Stress Corrosion Testing", WCAP-7628, non-proprietary, December, 1970.

5. Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water for PWR's".

FIGURE 1.
POST-LOCA RCS/SUMP BORON CONC VS PRE-TRIP RCS BORON CONC
D. C. COOK UNIT 1

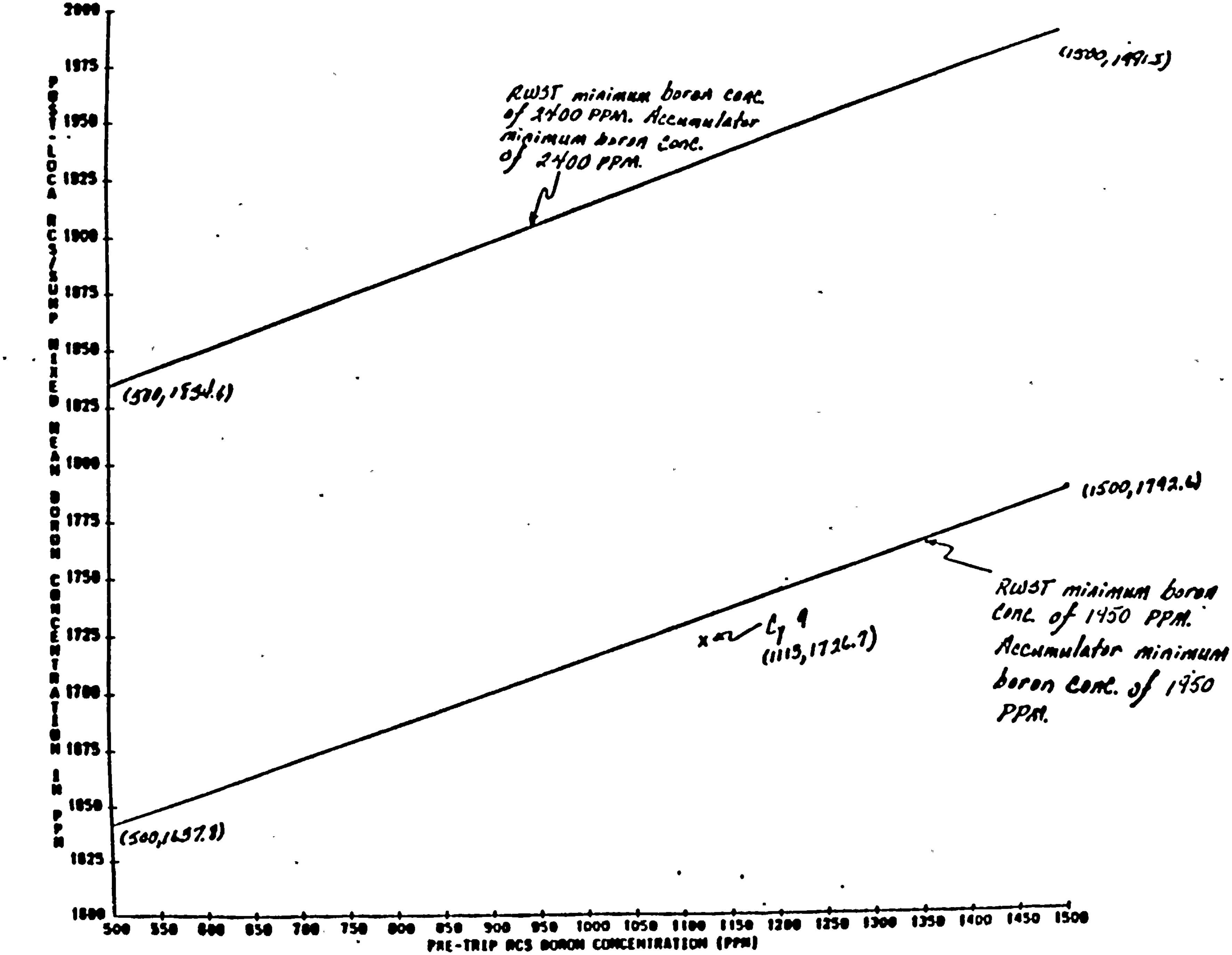


FIGURE 2.
POST-LOCA RCS/SUMP BORON CONC VS PRE-TRIP RCS BORON CONC
D. C. COOK UNIT 2

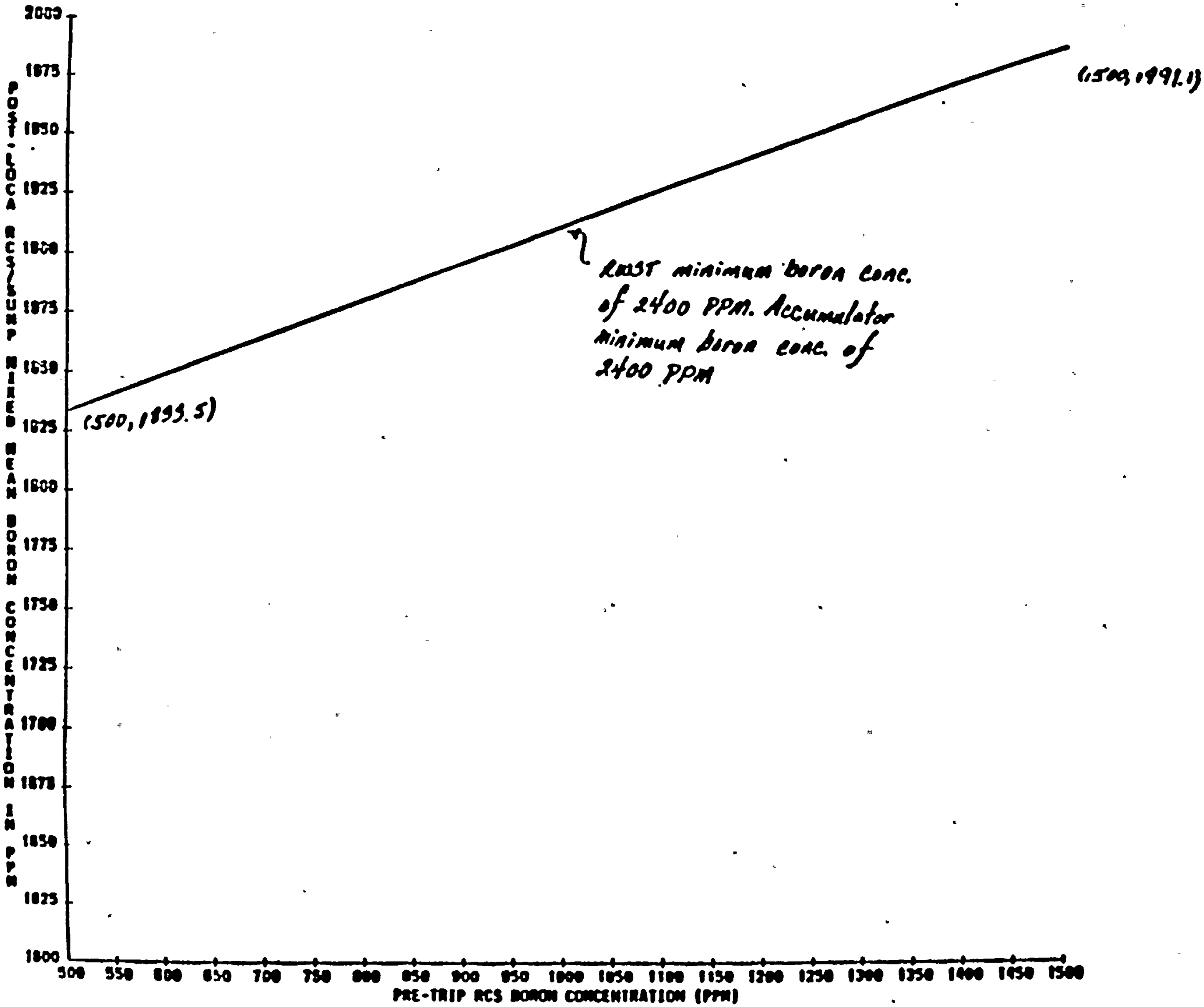
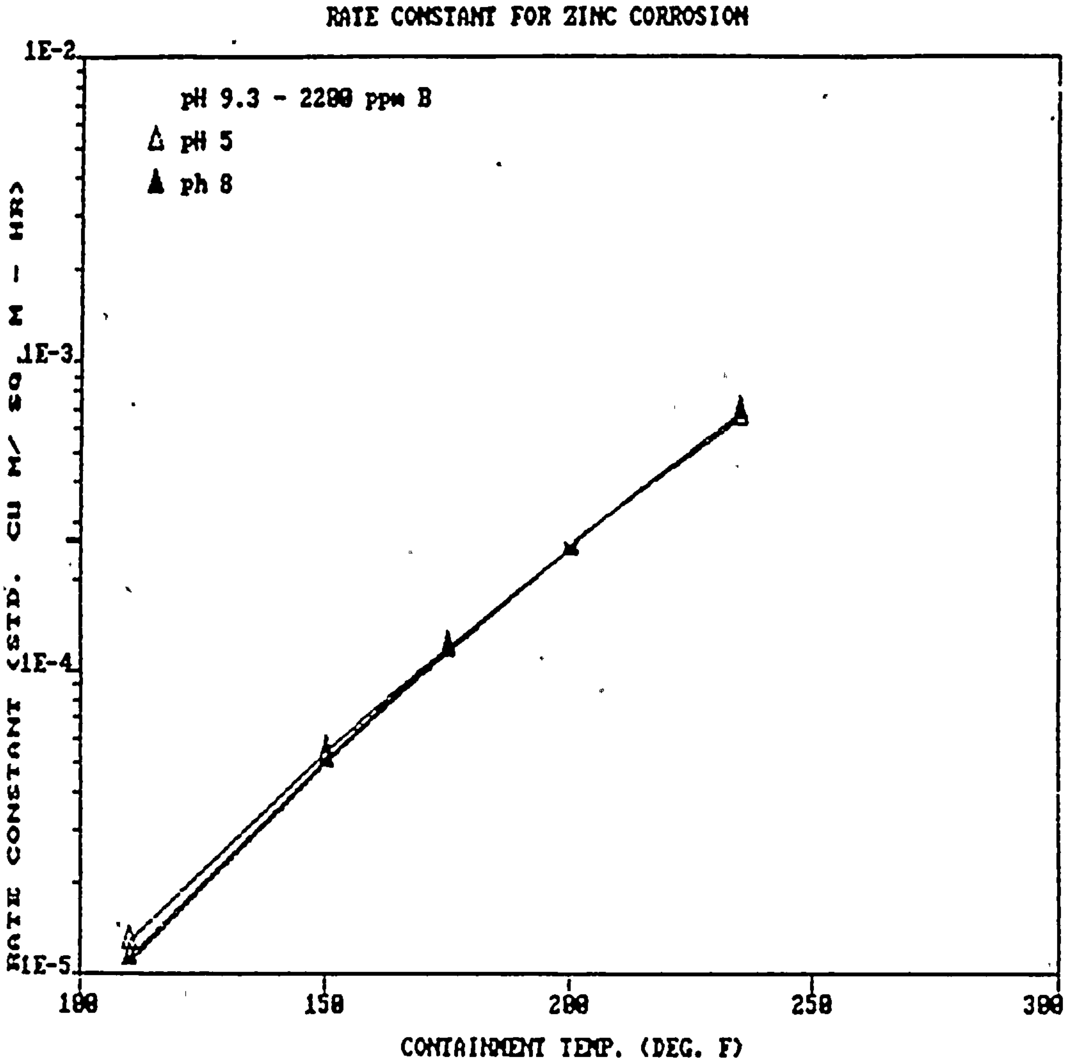


FIGURE 6-1

HYDROGEN PRODUCTION RATE CONSTANTS
FOR ZINC CORROSION



ATTACHMENT A
(Safety Evaluation)

FSAR/TECHNICAL SPECIFICATION

CHANGED PAGES

All active components of the safety injection system which operate during the injection phase of a loss of coolant accident are located outside the containment system. The safety injection pumps, centrifugal charging pumps, and residual heat removal pumps are located in the auxiliary building.

Recirculation Phase

Spilled coolant and injection water which is collected in the containment recirculation sump following the injection phase is recirculated back to the reactor coolant system by the residual heat removal pumps. The containment spray pump suction is also supplied directly from the containment recirculation sump. The reactor coolant system is supplied directly from the discharge of the residual heat removal heat exchangers, and from each of the heat exchanger outlets to the suction of the centrifugal charging and safety injection pumps which in turn pump into the coolant system.

The recirculation phase of operation has two modes, cold leg recirculation and hot leg recirculation. Initially, the discharge from the RHR pumps flows directly, and via the safety injection and charging pumps, to the same cold leg injection points used during the injection phase of operation. Later in recirculation, the discharge of each safety injection pump is, along with the RHR pump discharge, switched to two individual hot leg injection points. The switch to hot leg recirculation is made in order to mimimize the potential for boron precipitation.

Hot leg injection may begin during the recirculation phase of operation whenever the reactor coolant system and secondary coolant system are cooled down. The changeover to hot leg injection is specified to occur approximately 12 24 hours after the accident. At this time the residual heat generation rate has decayed to less than 1% of the nominal, the sensible heat in the steam generator secondary side will have been removed and the containment atmosphere and sump liquid temperature will have been reduced.

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 USABLE ~~contained~~ volume of ~~835~~ 2890 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 USABLE ~~contained~~ volume of ~~9690~~ 76,937 gallons,
2. A ~~minimum~~ boron concentration of ~~1950~~ 2400 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 water level volume of the tank, 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.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

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

1. A minimum ~~contained~~ USABLE volume of ~~5170~~ 5641 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 volume of 350,000 gallons of water,
2. A minimum boron concentration Between 2400 AND 2600 PPM, ~~of 1950 ppm,~~ and
3. A minimum solution temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between 929 and 971 cubic feet of borated water,

c. A ~~minimum~~ boron concentration between 2400 and 2600 ppm, ~~of 1950 PPM,~~ and

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

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig.

EMERGENCY CORE COOLING SYSTEMS

REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained volume of 350,000 gallons of borated water.

b. A ~~minimum~~ boron concentration of between 2400 AND 2600 ppm, ~~1950 ppm,~~ and

c. A minimum water temperature of 70°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water storage tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 70°F.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC) (Continued)

principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured and appropriately compensated MTC value is within the allowable tolerance of the predicted value provides additional assurances that the coefficient will be maintained within its limits during intervals between measurement.

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, and 3) T_{avg} is above the P-12 interlock setpoint.

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.

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 minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide ~~a~~ the Required SHUTDOWN MARGIN from all operating conditions ~~of 1.0% Δk/k~~ after xenon decay and cooldown to 200°F. The maximum boration capability requirement ~~occurs at EOL from full power equilibrium xenon conditions and~~ requires 5641 ~~5106~~ gallons of 20,000 ~~20,100~~ ppm borated water from the boric acid storage tanks or A USABLE VOLUME OF 99,598 ~~52,622~~ gallons of 2400 ~~1950~~ ppm borated water from the refueling water storage tank.

BASES

3/4.1.2 BORATION SYSTEMS (Continued)

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.

A usable volume of 2,890

The boron capability required below 200°F is sufficient to provide the Required ~~a~~ SHUTDOWN MARGIN ~~of 1% Δk/k~~ after xenon decay and cooldown from 200 to 140°F. This condition requires either ~~805~~ gallons of 20,000 ppm borated water from the boric acid storage tanks or ~~6000~~ gallons of ~~1950~~ 2400 ppm borated water from the refueling water storage tank.

A usable volume of 76,937

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis for a rod ejection accident.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} ... and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

EMERGENCY CORE COOLING SYSTEMS

BASES

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between ~~8.5~~ 7.6 and ~~11.0~~ 9.5 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 ECCS analyses to determine F_Q limits in Specifications 3.2.2 and 3.2.6 assumed a RWST water temperature of 70°F. The temperature value of the RWST water determines that of the spray water initially delivered to the containment following LOCA. It is one of the factors which determines the containment back-pressure in the ECCS analyses, performed in accordance with the provisions of 10 CFR 50.46 and Appendix K to 10 CFR 50.

ADVANCED NUCLEAR FUELS CORPORATION

600 108th AVENUE NE, PO BOX 90777, BELLEVUE WA 98009-0777
(206) 453-4300

FEB 0 6 1987

January 30, 1987
ANF-AEP/0550

Mr. Richard B. Bennett, Engineer
Nuclear Materials & Fuel Management
Indiana & Michigan Electric Company
c/o American Electric Power Service Corp.
One Riverside Plaza, 20th Floor
Columbus, OH 43215

Dear Mr. Bennett:

In response to your telephone request, Advanced Nuclear Fuels (ANF) has performed a review of the transient and LOCA analysis performed in support of D.C. Cook Unit 2. The review included the current work being performed for the steam line break and the analysis of record for the small break LOCA presented in the UFSAR. This review indicates that increasing the boron concentration in the refueling water storage tank (RWST) and the accumulators (ACC) to 2400 ppm would not adversely affect any of the ANF analysis.

If you have any questions regarding the above review, please feel free to contact our Mr. Jerry Holm (telephone 509-375-8142).

Sincerely,

L. D. O'Dell for

H. G. Shaw
Contract Administrator

gf

cc: M.P. Alexich
J.M. Cleveland
D.H. Malin
V. VanderBurg

Attachment 14 to AEP:NRC:0916W

SUMMARY OF ATTACHMENT 14

EVALUATION OF THE IMPACT OF 2000 GPM PRIMARY FLOW

ON THE UNIT 1 DILUTION TRANSIENT PERFORMED BY

WESTINGHOUSE ELECTRIC CORPORATION

RESPONSE TO QUESTION 9 OF REACTOR SYSTEMS BRANCH TRANSMITTAL

OF JANUARY 8, 1986

PREVIOUSLY SUBMITTED WITH AEP:NRC:0916P

LETTER, NS-TMA-2273, FROM T. M. ANDERSON (WESTINGHOUSE)

TO V. STELLO (NRC) DATED JULY 8, 1980

Summary of Attachment 14

This attachment is divided into three parts. The first part entitled, "Revision of Figure A-1 of NS-TMA-2273" describes a new analysis for D. C. Cook Unit 1 similar to that described in the letter from T. M. Anderson to V. Stello dated July 8, 1980 (Identifier NS-TMA-2273). The analysis was performed by our contractor, Westinghouse Electric Corporation. The curve from this calculation which corresponds to our maximum dilution flow rate of 225 gpm was used to prepare Unit 1 T/S Figure 3.1-3, Required Shutdown Margin.

The second part of this attachment is Attachment 1 to AEP:NRC:0916P. As indicated in Attachment 1 to AEP:NRC:0916P, the methodology of NS-TMA-2273 has been in use on Unit 1 since beginning of Cycle 6. Attachment 1 to AEP:NRC:0916P was approved in the SER for Amendment 82 to DPR-74.

The third part of this attachment is a copy of NS-TMA-2273. This document and Attachment 1 to AEP:NRC:0916P are being retransmitted to facilitate your review.

Revision of Figure A-1 of NS-TMA-2273

This discussion pertains to a revision of Figure A-1 provided in NS-TMA-2273. The scales of Figure A-1 have been extended to account for increased RCS boron concentrations Modes 4 and 5 (hot and cold shutdown). American Electric Power has indicated that the Mode 5 maintenance level minimum RHR flow rate is 2000 gpm. This is more limiting than the current minimum Mode 4 RHR flow rate of 3000 gpm. As such, the Mode 5 RHR flow rate of 2000 gpm was assumed for this revision. The maximum dilution flow rate is given as 225 gpm on page 14.C-21 (Unit 1) of D. C. Cook FSAR. The D. C. Cook Unit 1 plant specifics, as noted above, have been incorporated in the development of the revised curve.

American Electric Power has decided to incorporate in the Technical Specifications shutdown margin protection to ensure adequate operator response time for the mode 4 and 5 dilution transient. This is being done by applying the Westinghouse methodology described in NS-TMA-2273. In the process of generating a revised curve, which describes the shutdown margin requirements as a function of RCS boron concentration and possible dilution flow rate, certain assumptions of NS-TMA-2273 may no longer be applicable. In particular, the assumptions stating that in all cases a shutdown margin of 5% delta-k/k ($K_{eff} \leq 0.95$) is considered sufficient for continued operation without a requirement for control rod bank withdrawal is no longer valid. Due to increased RCS boron concentrations and the assumed minimum RHR flow rate of 2000 gpm, the revised curves show that a shutdown margin greater than 5% delta-k/k is required for dilution flow rates greater than 250 gpm.

Figure 1 provides the shutdown margin requirements as a function of initial Reactor Coolant System concentration and maximum possible dilution flow rate.

Figure 1 is based on D. C. Cook Unit 1 plant conditions as listed below:

1. The Reactor Coolant System effective volume is limited to the vessel and the active portions of the hot and cold legs when on RHR, i.e., steam generator volumes are not included.

2. The plant is borated to a shutdown margin greater than or equal to 1% delta-k/k.

3. Uniform mixing of clean and borated RCS water is not assumed, i.e., mixing of the clean, injected water and the affected loop is assumed but instantaneous, uniform mixing with the vessel, hot leg, and cold leg volume upstream of the charging lines is not assumed. Thus a "dilution front" moves through the cold legs, downcomer, and lower plenum to the core volume as a single volume front. This results in subsequent decreases in shutdown margin due to dilution fronts moving through the active core region with a time constant equal to the loop transit time when on RHR. The RHR flow rate assumed for this D. C. Cook Unit 1 figure is 2000 gpm.

Figure 1 notes areas of acceptable operation of different dilution flow rates as a function of the RCS boron concentration and borated shutdown margin (K_{eff}). For a given dilution flow rate, if the RCS boron concentration and shutdown margin result in a point placed to the left of the flow rate line, no control rod bank withdrawal is necessary. If the results place the plant to

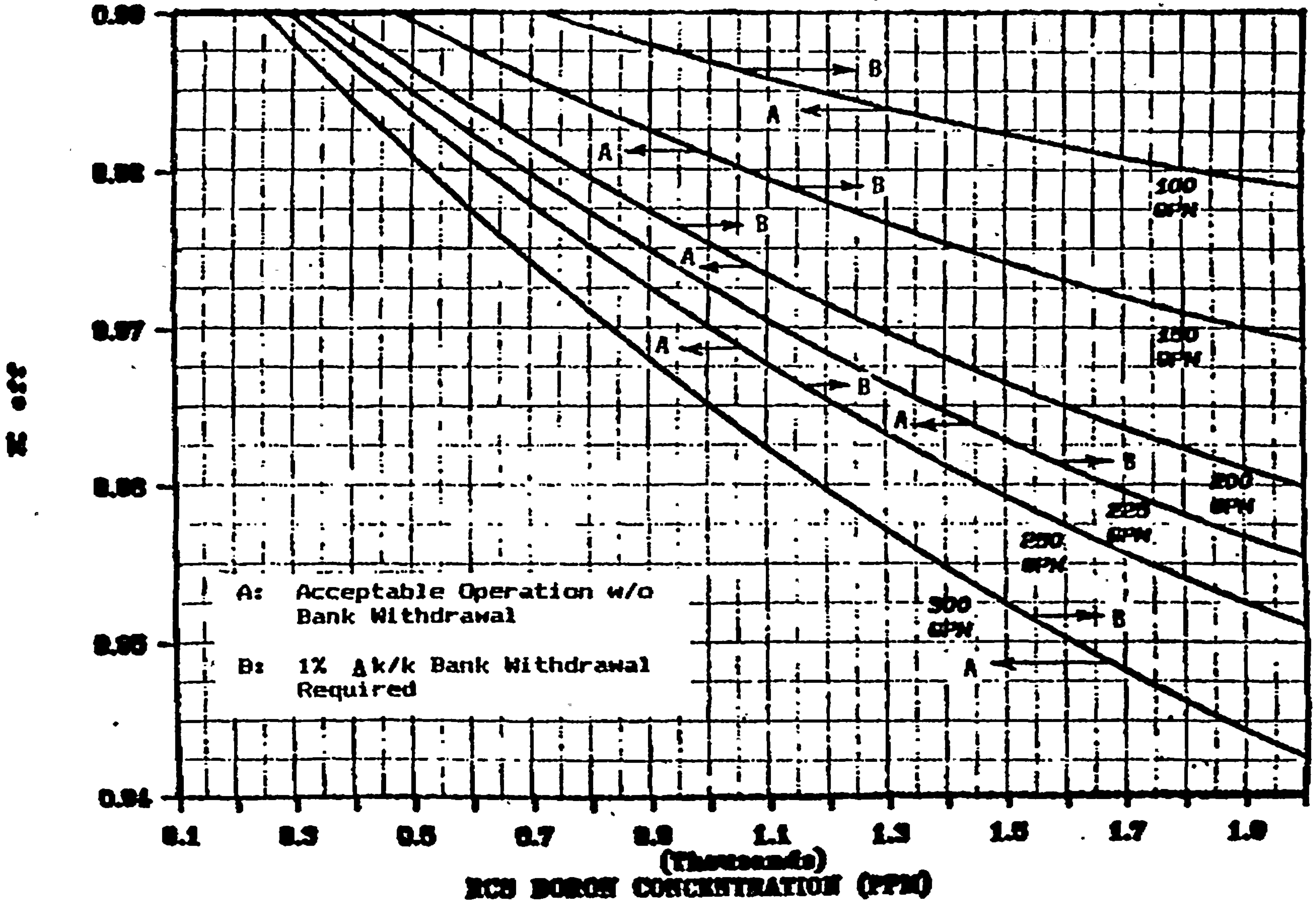
the right of the line, then either the shutdown margin must be increased such that the plant is moved to the area of acceptable operation, or 1% delta-k/k in control rods must be withdrawn to provide additional shutdown margin. The tripping of the withdrawn rods provides positive operator indication that a dilution event is in progress and additional time for operator termination of the event.

Figure 1 is based on best estimate calculations for the "all rods in" configuration.

Use of Figure 1 is applicable any time there is boration/dilution capability from the normal boric acid blending system. The above procedure is not required if boration and/or makeup during cold and hot shutdown is performed utilizing water from the RWST. This requires that the normal dilution/boration path is isolated from the charging path.

D. C. COOK UNIT 1

RHR FLOW RATE = 2800 GPM



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Question 9

The times required for loss of shutdown margin from boron dilution are provided on Page 188 of XN-NF-85-64. These times are significant for providing operating reaction times only following the initiation of an alarm. For each reactor condition given in Table 15.4.6.1, provide the time following initiation of the boron dilution event to the time when the alarm would function. Discuss diversity and redundancy of available alarms.

Response 9

A) The time from initiation of dilution to the time of alarm has not been specifically calculated for the analysis presented in XN-NF-85-64, Rev. 1, Supp. 1. Instead, the analysis in XN-NF-85-64 (P), Rev. 1, Supp. 1, was performed in a similar manner to the analysis presented in Section 14.1.5 of the Unit 2 Donald C. Cook Nuclear Plant Updated FSAR.

Additional detail on the FSAR analysis which bounds operation in Modes 4, 5 and 6 is provided in a letter (AEP:NRC:0860I) from M. P. Alexich to Harold R. Denton dated May 17, 1984. The analysis is also described in a letter (NS-TMA-2273) from T. M. Anderson of Westinghouse Electric Corporation to Victor Stello dated July 8, 1980. The results have been in use on Unit 1 since the beginning of Cycle 6 and on Unit 2 since the beginning of Cycle 3.

Both the FSAR analysis and the XN-NF-85-64(P), Rev. 1, Supp. 1 analysis for Modes 4, 5 and 6 ensure that 15 minutes are available from the initiation of dilution to the loss of shutdown margin. Volumes used in these analyses are limited to those assumed to have active flow.

As indicated in the updated FSAR and XN-NF-85-64(P), Rev. 1, Supp. 1, substantially longer times are available for operator response for the cases of dilution during startup and dilution during full power operation. The FSAR Mode 3 analysis is performed for startup from a reactor coolant system boron concentration of 2000 ppm.

B) Indications available to the operator include:

1) Status indication of the Chemical and Volume Control System and Reactor Makeup Water System with,

   a. Indication of boric acid and clean makeup flow rates including alarms on deviation from setpoint for both of these flows. These alarms would be expected to occur at the initiation of any inadvertent dilution involving the blender.

   b. CVCS valve position status lights, and

   c. Reactor Makeup Water Pump "running" status light.

Response 9 (Cont'd)

2) Source Range Neutron Flux with,

a. High Flux at Shutdown Alarm set at half a decade above background. This alarm is expected to occur after the dilution transient has been in progress for a period of time.

b. Use of the audible count rate indication to distinguish significant changes in flux, i.e., a doubling of the count rate.

c. Periodic, i.e., frequent surveillance of the Source Range meters and continuous strip chart recorder performed by the operator.

During startup operations, the high flux at shutdown alarm is not available. Additional indications available during startup operations include pressurizer and volume control tank levels. During power operations, the high flux at shutdown alarm and audible source range indications are not available. Source range meters and continuous strip chart indication are replaced by power range and intermediate range meters and a continuous strip chart which selectively displays these indications. When the rods are in automatic, rod insertion low and low-low alarms are available. When rods are in manual, Overtemperature Delta Temperature trip, alarm, and turbine runback are available.

.nghouse Electric Corporation Power Systems

PWR Systems Division

Box 355
Pittsburgh Pennsylvania 15230

July 8, 1980

NS-TMA-2273

Mr. Victor Stello
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, MD 20014

SUBJECT: Boron Dilution Concerns at Cold and Hot Shutdown

Dear Mr. Stello:

On June 27, 1980, Ed Jordan of your staff was notified by Westinghouse of an Unreviewed Safety Question under 10CFR50.59. This notification concerned the potential for an inadvertent boron dilution event while shutdown and operating on the Residual Heat Removal System. Attachment 1 is the text of the written notification supplied to our customers on July 8, 1980 which outlines potential Westinghouse concerns and the basis for recommended interim actions which address these concerns. These interim actions are somewhat modified from those previously reported. If there are any questions regarding the attached, please contact D. W. Call at 412/373-5074.

Very truly yours,

T. M. Anderson, Manager
Nuclear Safety Department

Attachment

cc: E. Jordan
R. Woods

ATTACHMENT 1

On June 27, 1980, you were notified of certain Westinghouse concerns and recommended actions regarding the potential for an inadvertent boron dilution event at cold or hot shutdown conditions while on the Residual Heat Removal System. This notification was in accord with Westinghouse determination that these concerns constitute an Unreviewed Safety Question under 10CFR Part 50.59. The NRC Office of Inspection and Enforcement was also notified on June 27, 1980 that these concerns have generic applicability to Westinghouse-supplied nuclear power plants. Further clarification was made to the NRC Office of Inspection and Enforcement on June 30, 1980 that Westinghouse concerns are not applicable while the plant is greater than 5% shutdown.

This letter is intended to formally document these concerns and to provide additional relevant information. This letter also modifies the earlier recommended actions by a more detailed specification of applicable plant operating conditions.

Inadvertent boron dilution at shutdown has been generally regarded as an event which can be identified and terminated by operator action prior to a return to critical. Automatic protection has not been a standard feature for Westinghouse plants. Westinghouse has recently been conducting a general investigation of this potential event relative to the licensing requirements imposed on newer plants not yet in operation. This investigation is not yet complete. However, it has been determined that under certain shutdown conditions and with certain assumed dilution rates, adequate time for operator action to prevent a return to critical may not be available.

The current Westinghouse evaluations are based on plant conditions as noted below:

1. The Reactor Coolant System effective volume is limited to the vessel and the active portions of the hot and cold legs when on RHR, i.e., steam generator volumes are not included.

2. The plant is borated to a shutdown margin greater than or equal to 1% $\Delta k/k$.

3. Uniform mixing of clean and borated RCS water is <u>not</u> assumed, i.e., mixing of the clean, injected water and the affected loop is assumed but instantaneous, uniform mixing with the vessel, hot legs, and cold leg volumes upstream of the charging lines is <u>not</u> assumed. Thus a "dilution front" moves through the cold legs, downcomer, and lower plenum to the core volume as a single volume front. This results in subsequent decreases in shutdown margin due to dilution fronts moving through the active core region with a time constant equal to the loop transit time when on RHR (five to seven minutes).

If a return to critical occurs as a result of an inadvertent dilution, the following potential concerns have been identified:

1. A rapid, uncontrolled power excursion into the low and intermediate power ranges occurs, resulting in a power/flow mismatch due to the low flow (approximately 1 - 2% of nominal) provided by the RHR pumps.

2. The potential exists for significant system overpressurization. Pressure increases above the RHR cut off head (approximately 600 psig) further accentuate the effects of a power/flow mismatch when all RCS (RHR) flow is lost. An investigation of the adequacy of existing cold overpressurization protection systems is necessary in order to assess the full impact of this potential problem.

3. The potential exists for limited fuel damage. This is not currently a significant concern. Preliminary evaluation indicates that the potential for exceeding DNB limits is low due to the cold initial operating conditions. Further investigation of this problem is underway.

The recommended interim actions to prevent or mitigate an inadvertent boron dilution at shutdown conditions are detailed in Appendix A. If no cocked control rods are required, as specified in Figure A-1, the plant operator has fifteen minutes from the initiation of dilution event to terminate the event before a return to critical occurs. It is the Westinghouse position that a fifteen minute time interval from the initiation of the dilution to the time shutdown margin is lost is sufficient time for operator action. If cocked control rods are required, the source range reactor trip provides positive indication for immediate operator action to terminate dilution.

It is expected that the operator has available the following information for determination that a dilution event is in progress:

1. Source Range Neutron Flux with,

   a. High Flux at Shutdown Alarm set at half a decade above background.

   b. Use of the audible count rate indication to distinguish significant changes in flux, i.e., a doubling of the count rate.

   c. Periodic, i.e., frequent surveillance of the Source Range meters performed by the operator.

2. Status indication of the Chemical and Volume Control System and Reactor Makeup Water System with,

a. Indication of boric acid and blended (total) flow rate, or

b. Indication of boric acid and clean makeup flow rate,

c. CVCS valve position status lights, and

d. Reactor Makeup Water Pump "running" status light.

The operator action necessary upon determination that a dilution event is in progress (by High Flux at Shutdown Alarm, Source Range Reactor Trip, "P-6 Available" indication, high indicated or audible count rates, or make up flow deviation alarms) is:

1. Immediately open the charging/SI pump suction valves from the RWST (that open on receipt of an "S" signal). (For 312 plants these are LCV-115-B, D. For 412 plants these are LCV-112-D, E.)

2. Immediately close the charging/SI pump suction valves from the VCT (that close on receipt of an "S" signal). (For 312 plants these are LCV-115-C, E. For 412 plants these are LCV-112-B, C.)

3. For two-loop plants, immediately open the charging suction valves from the RWST. (For 212 plants these are LCV-113-B and LCV-112-C.) Also immediately close the charging suction valves from the VCT. (For 212 plants these are LCV-113-A and LCV-112-B.)

Through the use of Appendix A and the above noted operator action requirements, Westinghouse is attempting to minimize the operational burden placed on the plant to prevent or mitigate an inadvertent dilution event while maintaining adequate safety margin. Our investigation of this event is continuing. A detailed analytical model of the system response to a dilution event at shutdown conditions is being developed and the potential for system overpressurization and fuel failure will subsequently be assessed. The Westinghouse investigation is expected to be completed by September 15, 1980. We will keep you informed as to the results of our efforts.

APPENDIX A

Figure A-1, attached, provides the shutdown margin requirements as a function of Reactor Coolant System boron concentration and maximum possible dilution flow rate. Prior to use of this figure, the plant must determine the maximum dilution flow rate of all charging pumps not rendered inoperable once the plant is placed on RHR. To cover all modes, it should be assumed that the flow rate is based on pump runout unless there are flow limiting devices in the system (orifices, piping resistances, etc.). The Reactor Makeup Water pump capacity may be limiting in the determination of the maximum possible dilution flow rate.

Figure A-1 notes areas of acceptable operation of three different dilution flow rates as a function of RCS boron concentration and borated shutdown margin (K_{eff}). For a given dilution flow rate, if the RCS boron concentration and shutdown margin result in a point placed to the left of the flow rate line, no control rod bank withdrawal is necessary. If the results place the plant to the right of the line, then either the shutdown margin must be increased such that the plant is moded to the area of acceptable operation, or 1% $\Delta k/k$ in control rods must be withdrawn to provide additional shutdown margin. The tripping of the withdrawn rods provides positive operator indication that a dilution event is in progress and additional time for operator termination of the event. In all cases, a shutdown margin of 5% $\Delta k/k$ ($K_{eff} \leq 0.95$) is considered sufficient for continued operation without a requirement for control rod bank withdrawal.

Figure A-1 is based on best estimate calculations for the "all rods in" configuration. It is recommended that the Westinghouse Nuclear Design Report for your plant be used as a reference in determining the RCS boron concentration with the appropriate conservatism to be used in the figure. The Westinghouse Nuclear Fuel Division is available to provide assistance in meeting the constraints imposed by the Figure A-1 requirements.

Use of Figure A-1 is applicable any time there is boration/dilution capability from the normal boric acid blending system. The above procedure is not required if boration and/or makeup during cold and hot shutdown is performed utilizing water from the RWST. This requires that the normal dilution/boration path is isolated from the charging path. Two means of lockout to isolate the charging path are available:

1. Lock out Reactor Makeup Water Supply.

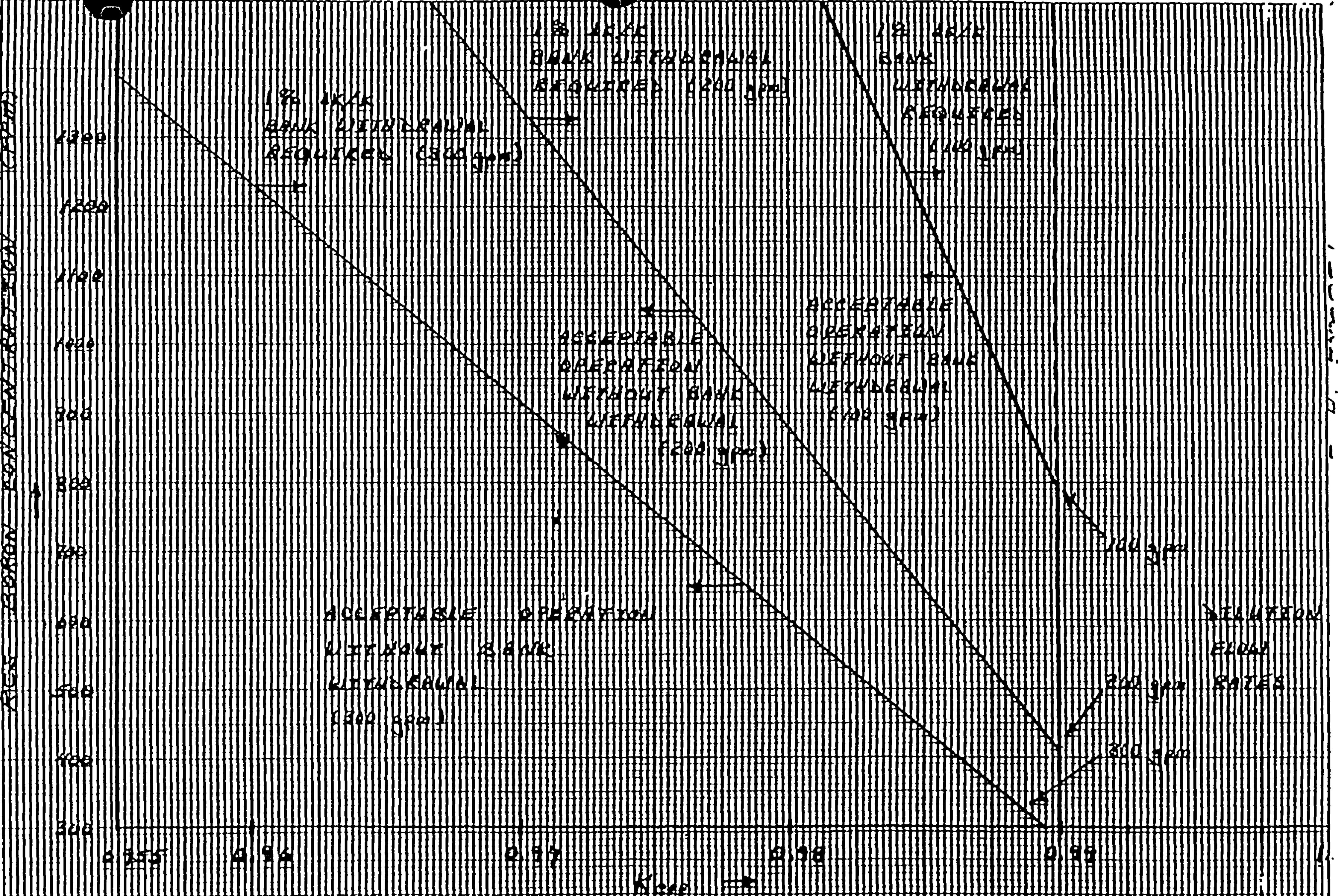
   This is accomplished by valve 8338 for 212 plants, valve 8457 for 312 plants, and valve 8455 for 412 plants.

OR:

2. Lock out valves between the boric acid blender and the VCT.

   These are FCV-111B, FCV-110B, 8339, 8355, and 8361 for 212 plants; FCV-114A, FCV-113B, 8454, 8441, and 8439 for 312 plants; FCV-111B, FCV-110B, 8453, 8441, 8439 for 412 plants.

This recommendation precludes the occurrence of an inadvertent dilution while borating or making up water from the RWST under these conditions.



SHUTDOWN MARGIN REQUIREMENTS AS A FUNCTION OF RCS BORON CONCENTRATION AND DILUTION FLOW RATE

ADVANCED NUCLEAR FUELS CORPORATION

600 108th AVENUE NE, PO BOX 90777, BELLEVUE, WA 98009-0777
(206) 453-4300

MAR 11 1987

March 5, 1987
ENC/AEP-0556

Mr. Rick Bennett, Engineer
Nuclear Materials & Fuel Management
Indiana & Michigan Electric Company
c/o American Electric Power Service Corp.
One Riverside Plaza, 20th Floor
Columbus, OH 43216-6631

Dear Mr. Bennett:

Attached is a recommended change to the D.C. Cook Unit 1 Technical Specification on F_Q to allow operation of ANF fuel to peak pellet exposures of 51 GWd/MT. A justification of this change is also attached for your use in obtaining NRC approval for this change. This is a revision to our letter ENC/AEP-0535 dated November 11, 1986.

If you have any questions regarding the attachment, please contact our Mr. J.S. Holm (telephone 509-375-8142).

Sincerely,

JS Holm for

H. G. Shaw
Contract Administrator

gf

Attachment

cc: J. M. Cleveland
D. H. Malin
V. VanderBurg
J. S. Holm (ANF)

Attachment

D.C. COOK UNIT 1 TECHNICAL SPECIFICATION CHANGE

Ref: (1) XN-NF-85-115, Rev. 1, "D.C. Cook Unit 1 Limiting Break K(Z) LOCA/ECCS Analysis," November 1986.

(2) XN-NF-85-68(P), Rev. 1, "Donald C. Cook Unit 2 Limiting Break LOCA/ECCS Analysis, 10% Steam Generator Tube Plugging, and K(Z) Curve," April 1986.

(3) XN-NF-85-117, Supp. 1, "St. Lucie Unit 1 Revised LOCA/ECCS Analysis with 15% Steam Generator Tube Plugging Break Spectrum and Exposure Results," December 1985.

A LOCA/ECCS analysis justifying the operation of ANF fuel currently in the D.C. Cook Unit 1 reactor is presented in Reference 1. The analysis in that report supports a peak F_Q of 2.04 with an axial dependence as shown in Figure 1. This analysis is applicable to the ANF fuel currently in the D.C. Cook Unit 1 reactor, with a minimum peak rod average exposure greater than 20 GWd/MT and anticipated to be less than 47 GWd/MT.

Justification for an exposure independent F_Q for D.C. Cook Unit 1 is based on an exposure analysis for D.C. Cook Unit 2 (Reference 2). Peak cladding temperatures are dependent upon fuel rod initial stored energy, which for the EXEM/PWR models increases from 0 to about 2 GWd/MTM and then decreases with exposure. The analysis for D.C. Cook Unit 2 with 17x17 fuel geometry demonstrated that over the exposure range of 0 to 47 GWd/MTM, the peak cladding temperature decreased with exposure for exposures beyond the peak stored energy exposure. A similar trend was observed for St. Lucie Unit 1 with 15x15 fuel geometry (Reference 3). Similar results would be expected for D.C. Cook Unit 1 with 15x15 fuel geometry using EXEM/PWR models. Based on the trend of decreasing peak cladding temperature with increasing exposure, the analysis in Reference 1 is conservative and supports an exposure independent F_Q of 2.04, along with the K(Z) curve shown in Figure 1, for ANF fuel at peak rod average exposures between 20 and 47 GWd/MTM. A peak rod average exposure of 47 GWd/MTM is equivalent to a peak pellet exposure of 51 GWd/MTM.

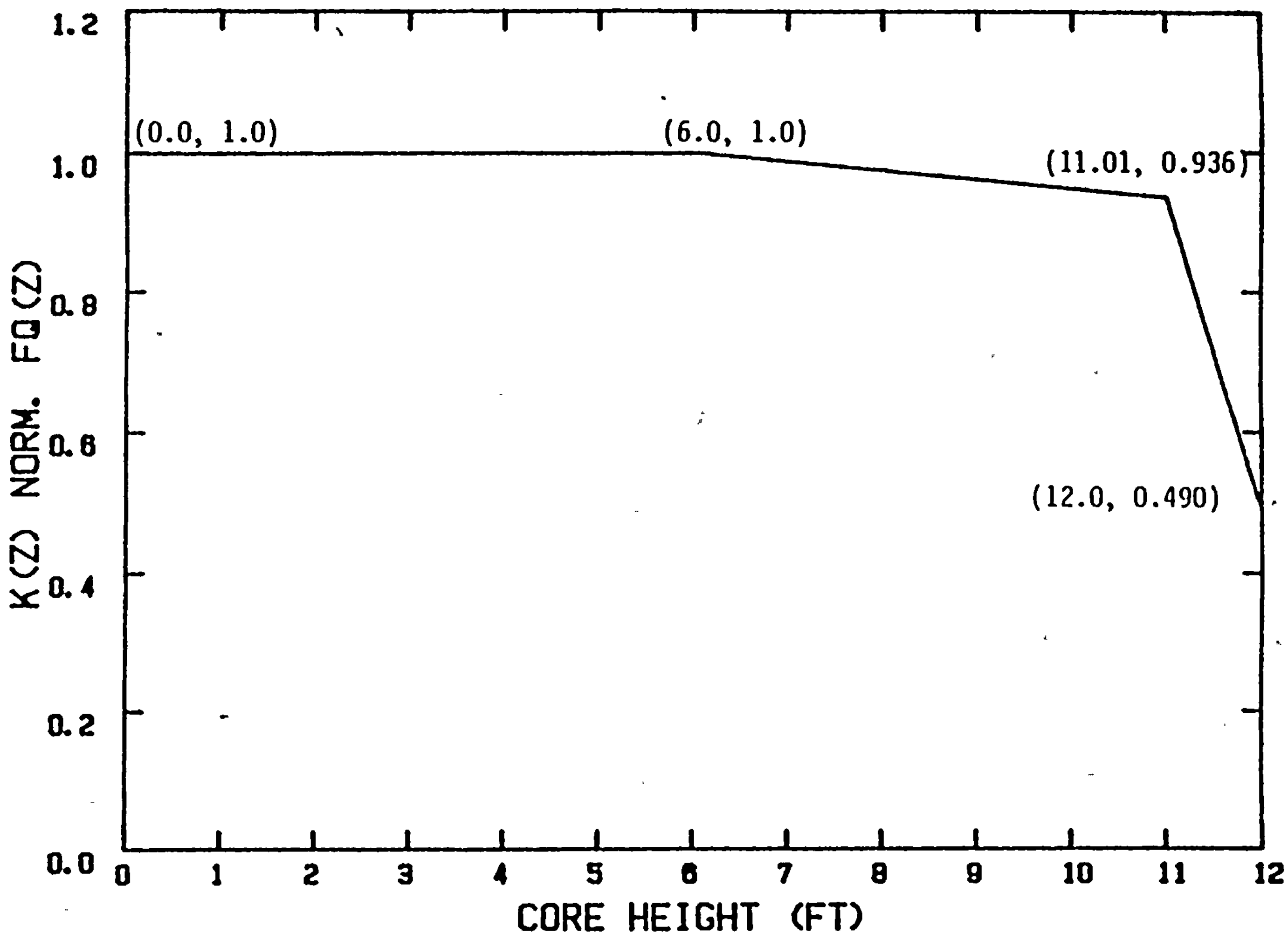


Figure 1 Hot Channel Factor Normalized Operating Envelope, F_Q=2.04, K(Z) Function

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

January 29, 1987
AEP:NRC:0940E

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
License No. DPR-58
D. C. COOK UNIT 1 LIMITING BREAK K(Z)
LOCA/ECCS ANALYSIS

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Dear Sirs:

The purpose of this letter is to inform you that Exxon Nuclear Company (ENC) has transmitted to you proprietary and non-proprietary copies of their report No. XN-NF-85-115, Rev. 2, entitled "D. C. Cook Unit 1 Limiting Break K(Z) LOCA/ECCS Analysis," via their letter No. GNW:001:87, dated January 15, 1987. By this letter, we request that these documents be added to our Unit 1 docket, No. 50-315. The report documents the results of the LOCA/ECCS analysis performed by ENC to determine the K(Z) for the ENC fuel in Unit 1 of the D. C. Cook Plant. The analysis supports operation of D. C. Cook Unit 1 at its currently licensed thermal power rating of 3250 MW.

Revision 0 of this report was transmitted to you on February 5, 1986 via our letter AEP:NRC:0940C. In that letter, (and in the NRC staff's subsequent safety evaluation report dated February 21, 1986) it was indicated that the Fuel Cooling Test Facility (FCTF) reflood correlations which were used by ENC in their analysis were undergoing NRC review, and that the K(Z) curve presented in XN-NF-85-115 Rev. 0 would be reexamined after completion of the NRC's review of the FCTF data. Subsequent to this, ENC has modified the FCTF correlations to resolve NRC concerns. ENC has received formal approval from the NRC to use the correlations as modified. The analyses presented in Revision 2 to XN-NF-85-115 utilize the revised FCTF correlations. We note, however, that the K(Z) curve presented in Revision 0 of XN-NF-85-115 remains unchanged in Revision 2 to that document. Revision 2 to XN-NF-85-115 also incorporates minor editorial changes to Table 2.1 of the document. These changes correct errors in the listed volumes for the reactor vessel and pressurizer and add a footnote to denote the amount of steam generator tube plugging used in the analysis. These changes are editorial only, and do not impact the K(Z) results.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President 1/29/87

Attachment

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

February 20, 1987
AEP:NRC:1018

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
License No. DPR-58
PROPOSED TECHNICAL SPECIFICATION CHANGE REGARDING
EXTENSION OF PEAK PELLET EXPOSURE FOR
ADVANCED NUCLEAR FUEL CORPORATION FUEL

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Dear Sirs:

This letter and its attachments constitute an application for amendment to the Technical Specifications (T/Ss) for the Donald C. Cook Nuclear Plant Unit No. 1. Specifically, we propose to modify T/Ss 3/4.2.2 (Heat Flux Hot Channel Factor - $F_Q(Z)$) and 3/4.2.6 (Axial Power Distribution) to allow an increase in the allowed peak pellet exposure for Advanced Nuclear Fuel Corporation (ANF) (formerly Exxon Nuclear Company) fuel from its present value of 48.0 MwD/kg to 51.0 MwD/kg.

Predictions of fuel burnup made prior to the beginning of the current cycle indicated that the ANF fuel would not exceed the current peak pellet exposure limit of 48.0 MwD/kg. However, recent flux maps have indicated the potential for the ANF fuel to slightly exceed the limit prior to discharge at the end of cycle. According to our flux maps, the limit may be exceeded as early as May 3, 1987, approximately three weeks prior to the start of the upcoming Unit 1 refueling outage, currently scheduled to begin on May 24, 1987. Because this situation creates the potential for a required early shutdown of the unit, we request an expedited review of the proposed changes and a response by April 30, 1987. We are currently preparing proposed simplifications to the D. C. Cook Unit 1 power distribution monitoring T/Ss. These proposed changes, which are intended to provide consistency between the D. C. Cook Units 1 and 2 T/Ss, will most likely propose deletion of the burnup requirements from the T/Ss. However, because we will reach our peak pellet exposure limit in early May 1987, we have decided to submit the peak pellet exposure extension request separately to allow adequate time for NRC review.

The reasons for the proposed changes and our analysis concerning significant hazards considerations are contained in Attachment 1 to this letter. The proposed revised T/S pages are contained in Attachment 2. Attachments 3 and 4 contain evaluations performed by ANF in support of the changes. These evaluations are discussed in more detail in Attachment 1.

Since Attachment 4 contains ANF proprietary information, we have included an affidavit to that effect with it. Attachment 5 contains a non-proprietary version of the ANF document in Attachment 4.

The ANF analyses we have attached provide justification for an extension of the allowed peak pellet exposure for their fuel to 48.7 Mwd/kg, rather than the 51.0 Mwd/kg we have proposed in this submittal. As detailed in Attachment 1, it is our understanding that the additional analyses necessary to support the value of 51.0 Mwd/kg can be reviewed by us under the provisions of 10 CFR 50.59 and therefore will not require an additional submittal. The value of 48.7 Mwd/kg should be sufficient to allow operation to continue until the start of the Unit 1 refueling outage, currently scheduled for May 24, 1987. At this time, however, we are investigating the possibility of delaying the outage start date due to various system concerns, such as outages in other of our operating units. For this reason, we are considering having analyses performed to justify peak pellet exposure limits for ANF fuel greater than 48.7 Mwd/kg. ANF has informed us that these analyses may be extensive and involve several weeks preparation time. In order to allow adequate time for NRC review of our proposed changes and for our own evaluation of our peak pellet exposure needs, we have chosen to submit analyses supporting peak pellet exposures of 48.7 Mwd/kg and to pursue exposures beyond this value via the 10 CFR 50.59 process. This approach was discussed with the NRC staff on February 12, 1987. Since at the present time we can only justify a value of 48.7 Mwd/kg, we would implement administration controls to prohibit operation above peak pellet exposures for ANF fuel of 48.7 Mwd/kg without appropriate analyses and 10 CFR 50.59 review.

We believe that the proposed changes will not result in (1) a significant change in the types of effluents or a significant increase in the amounts of any effluents that may be released offsite, or (2) a significant increase in individual or cumulative occupational radiation exposure.

These proposed changes have been reviewed by the Plant Nuclear Safety Review Committee (PNSRC), and will be reviewed by the Nuclear Safety and Design Review Committee (NSDRC) at their next regularly scheduled meeting.

In compliance with the requirements of 10 CFR 50.91(b)(1), copies of this letter and its attachments have been transmitted to Mr. R. C. Callen of the Michigan Public Service Commission and Mr. G. Bruchmann of the Michigan Department of Public Health.

Pursuant to 10 CFR 170.12(c), we have enclosed an application fee of $150.00 for the proposed amendment.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to insure its accuracy and completeness prior to signature by the undersigned.

Very truly yours,

M. P. Alexich
Vice President

cm

Attachments

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
G. Bruchmann
R. C. Callen
G. Charnoff
NRC Resident Inspector - Bridgman
J. G. Keppler - Region III

Attachment 1 to AEP:NRC:1018

Reasons and 10 CFR 50.92
Analyses for Changes to the
Donald C. Cook Nuclear Plant Unit No. 1
Technical Specifications

Background

This letter proposes to increase the allowable peak pellet exposure for ANF fuel from its present value of 48.0 Mwd/kg to a higher value of 51.0 Mwd/kg. Peak pellet exposure is in general limited by either LOCA analysis considerations or fuel mechanical design characteristics. For ANF fuel in Unit 1, the value has been included in the T/Ss specifically because of LOCA analysis considerations, which are discussed in more detail below. The limit of 48.0 Mwd/kg appears in the graphs of exposure-dependent F_Q limit (F_Q^L (E ℓ)) and normalized F_Q limit (T (E ℓ)) found in Figure 3.2-4 of the Unit 1 T/Ss (page 3/4 2-23). It also appears in the F_Q uncertainty factors E_p(Z) (page 3/4 2-7) and F_p (page 3/4 2-20).

During the design phase of a fuel cycle, predictions of peak pellet exposure are made, and these predicted exposures are ensured to be within applicable limits (mechanical and LOCA, as well as T/S where applicable). For ANF assemblies in D. C. Cook Unit 1, we monitor burnup via flux mapping to ensure adherence to T/S limits. Recent flux mapping has demonstrated that the potential exists for several ANF fuel assemblies to slightly exceed their 48.0 Mwd/kg T/S limit by May 3, 1987, approximately three weeks prior to the scheduled start of the upcoming Unit 1 outage, which is currently scheduled to begin on May 24, 1987.

Currently, all new fuel for D. C. Cook Unit 1 is being supplied by Westinghouse Electric Corporation (Westinghouse). The present cycle (Cycle 9) uses only 34 ANF assemblies. Of these 34 assemblies, only 4 are expected to exceed the current peak pellet exposure limit of 48.0 Mwd/kg. By May 24, 1987 none should have exceeded the limit by more than 0.7 Mwd/kg, which represents an excess of less than 2%. Current design plans for the Cycle 10 core do not call for any of the ANF assemblies to be reused, although these plans are subject to change should we encounter unanticipated fuel failures or damaged assemblies during refueling.

ANF has evaluated the safety impact of operation up to 51.0 Mwd/kg for LOCA considerations (Attachment 3), but only to 48.7 Mwd/kg for mechanical design

considerations (Attachment 4). The mechanical design evaluation was limited to 48.7 Mwd/kg because this value could be supported in large part by extrapolations from existing analyses. ANF has informed us that analyses to support higher values of peak pellet exposure may be extensive and involve several weeks preparation time. Thus, we were unable to have these analyses performed in time to accompany this letter and still allow adequate time for NRC review. Additionally, as discussed in the cover letter, we are unsure at this time whether peak pellet exposures beyond 48.7 Mwd/kg are even necessary. We are therefore unsure whether we want to undertake the expense and effort to have the analyses performed.

ANF has informed us that in meetings with the NRC staff held in August 1986, the staff explained that fuel mechanical design analyses could be reviewed under the provisions of 10 CFR 50.59 without NRC review provided that ANF followed the methodology outlined in their document XN-NF-82-06, Rev. 1, "Qualification of Exxon Nuclear Fuel for Extended Burnup", and if the batch average is below the approved high burnup level in this document. Should we decide to pursue peak pellet exposures beyond 48.7 Mwd/kg, which equates to batch average burnup considerably less than batch average burnups approved in XN-NF-82-06 Rev. 1, we propose to have ANF do so using the parts of XN-NF-82-06 Rev. 1 which are applicable to peak pellet exposure, and to review these analyses under the provisions of 10 CFR 50.59. (Since peak rod and peak assembly exposures are not being changed beyond that addressed in the currently approved mechanical design safety evaluation, XN-NF-84-25, not all aspects of the XN-NF-82-06 Rev. 1 methodology need to be addressed.)

Description of Proposed Changes

The ANF evaluations presented in Attachments 3 and 4 provide support for a peak pellet exposure limit of 51.0 Mwd/kg based on LOCA considerations, but only 48.7 Mwd/kg based on mechanical design considerations. These analyses allow the exposure-dependent peaking factor limit, $F_Q^L(E\ell)$ of T/S Figure 3.2-4 (p. 3/4 3-23) to remain at 1.82 (its present value at 48.0 Mwd/kg peak pellet exposure). We have redrawn T/S Figure 3.2-4 to show the curve extending to an $F_Q^L(E\ell)$ value of 1.82 at 51.0 Mwd/kg. T $(E\ell)$, the normalized $F_Q^L(E\ell)$, which is also contained in T/S Figure 3.2-4, has been

similarly redrawn. We have also modified the values of $E_p(Z)$ in T/S 4.2.2.2 (p. 3/4 2-7) and F_p in T/S 3.2.6.g. (p. 3/4 2-20) to define these factors as 1.0 from 48.0 to 51.0 Mwd/kg peak pellet exposure. $E_p(Z)$ is an uncertainty factor to account for a reduction in the $F_Q^L(E\ell)$ curve due to an accumulation of exposure between flux maps. The quantity F_p is a similar factor for use with the Axial Power Distribution Monitoring System (APDMS). The values of these factors are related to the slope of the $F_Q^L(E\ell)$ curve from T/S Figure 3.2-4. A flat slope for the $F_Q^L(E\ell)$ curve, as we have proposed between 48.0 and 51.0 Mwd/kg, results in no change in the allowable value of $F_Q^L(E\ell)$ between flux maps and thus no penalty (penalty factor of 1.0). This is consistent with the value of 1.0 assigned to these factors between peak pellet exposures of 0.0 and 17.62 Mwd/kg where the slope of $F_Q^L(E\ell)$ is also flat. Since at the present time we can only justify a peak pellet exposure of 48.7 Mwd/kg, we would implement administrative controls to prohibit operation beyond 48.7 Mwd/kg without an analysis which uses the methodology from the appropriate sections of XN-NF-82-06 Rev. 1 and a subsequent review of these analyses under 10 CFR 50.59.

Justification for Proposed Changes

The following justifications address LOCA considerations up to 51.0 Mwd/kg and mechanical design considerations up to 48.7 Mwd/kg. As discussed previously, we propose that any additional mechanical design analyses which may be performed in support of higher peak pellet burnups will be performed using the approved methodology of XN-NF-82-06 Rev. 1 and will be reviewed under the provisions of 10 CFR 50.59.

1. LOCA Considerations

   F_Q does not vary as a function of burnup for Westinghouse fuel in either the D. C. Cook Units 1 or 2 T/Ss. For ANF fuel, it varies as a function of burnup only in the Unit 1 T/Ss. The reason the burnup dependence is included for ANF fuel in Unit 1 is that the limits were based on ANF LOCA analyses dating back to the mid-1970s, which used a burnup-dependent F_Q. More detailed and modern ANF LOCA analyses do not require F_Q to be burnup-dependent. For example, F_Q for ANF fuel in

D. C. Cook Unit 2 is a constant at 2.10, with no exposure dependence or limits found in the T/Ss. The newer ANF analyses have determined the limiting exposures with regard to peak clad temperature concerns to be at relatively low exposures (less than 10 Mwd/kg). Similarly, Westinghouse LOCA models assume a constant value for F_Q throughout the cycle.

ANF has recently performed a new limiting break K(Z) LOCA/ECCS analysis for Unit 1. This analysis, which is contained in XN-NF-85-115 Rev. 2, was sent to you directly by ANF in their letter GNW:001:87, dated January 15, 1987 (as noted in our letter AEP:NRC:0940E, dated January 29, 1987). This analysis used the modern ANF evaluation methods including the Fuel Cooling Test Facility (FCTF) reflood heat transfer correlations. The document discusses analyses performed for peak pellet exposures of 2 Mwd/kg and 9 Mwd/kg, which ANF has determined to be bounding with regard to peak clad temperature. These analyses assumed an F_Q value of 2.04 peaked at the core midplane at 2 Mwd/kg and 1.95 peaked at the core top at 9 Mwd/kg. Both of these values are conservative with respect to the value of 1.82 required by Unit 1 T/S Figure 3.2-4 at 48 Mwd/kg.

As discussed in Attachment 3, ANF has informed us that the analyses they performed for XN-NF-85-115 Rev. 2 are applicable up to a peak rod average exposure of 47 Mwd/kg, which corresponds to a peak pellet exposure of 51 Mwd/kg. This is based on comparisons of exposure analyses ANF performed for their fuel in D. C. Cook Unit 2 and St. Lucie Unit 1. The analyses for both of these units demonstrated maximum values of peak clad temperature occurring in the very low exposure range. For D. C. Cook Unit 2, the peak temperature occurred at an exposure of only 2 Mwd/kg. Since all the ANF assemblies have undergone significant burnup, we did not need an F_Q value as high as that supported by the ANF analyses and have thus conservatively proposed to maintain F_Q at a value of 1.82, which corresponds to its present limit at 48.0 Mwd/kg.

2. Mechanical Design Considerations

The analysis supporting the current peak pellet exposure of 48.0 Mwd/kg is contained in ANF report XN-NF-84-25 (P), entitled "Mechanical Design Report Supplement for D. C. Cook Unit 1 Extended Burnup Fuel Assemblies." This document was submitted directly to you by ANF with their letter JCC:113:84, dated August 21, 1984. It was referenced by us in our letter AEP:NRC:0745M, dated August 23, 1984, which proposed to increase peak pellet exposure for ANF fuel in D. C. Cook Unit 1 from 42.2 Mwd/kg to its present value of 48.0 Mwd/kg. The changes were approved by the NRC via Amendment 82 to the D. C. Cook Unit 1 T/Ss, which is dated November 29, 1984.

Attachment 4 to this letter contains an evaluation by ANF to support extending the peak pellet burnup to 48.7 Mwd/kg. This evaluation demonstrates that applicable ANF mechanical design criteria would be satisfied with a peak pellet exposure limit of 48.7 Mwd/kg.

Of these criteria, which are discussed in Attachment 4, ANF has determined that all criteria except steady-state strain, corrosion, hydrogen absorption, and fuel rod internal pressure are essentially independent of the peak pellet exposure limit. For steady-state strain, corrosion, and hydrogen absorption, ANF performed extrapolations of their analyses reported in XN-NF-84-25 (P). The results of these extrapolations, reported in Attachment 4, demonstrate significant margin to the ANF design limits. For fuel rod internal pressure, ANF performed a new analysis using their RODEX2 code. The peaking factor was increased by 2% at the maximum axial region from that used for the XN-NF-84-25 analysis to bound the increased peak pellet burnup. The results of this analysis demonstrated a peak internal pressure well below the ANF design criteria limit of 2250 psia specified in XN-NF-84-25.

Significant Hazards Considerations

Per 10 CFR 50.92, a proposed amendment will not involve a significant hazards consideration if the proposed amendment does not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated,

(2) create the possibility of a new or different kind of accident from any accident previously evaluated, or

(3) involve a significant reduction in a margin of safety.

Criterion 1

We have presented analyses which demonstrate that operation up to 48.7 Mwd/kg peak pellet exposure will not violate any applicable safety limits or design criteria. In addition, we would implement administrative controls to prohibit operation beyond 48.7 Mwd/kg unless analyses are performed using methodology that is known to be acceptable to the NRC. Therefore, we conclude that the proposed changes will not significantly increase the probability of occurrence or consequences of a previously evaluated accident, nor will they involve a significant reduction in a margin of safety.

Criterion 2

LOCA analyses and fuel mechanical design limits are the principal areas of concern regarding peak pellet exposure. We have presented evaluations which conclude that applicable criteria with regard to these issues will continue to be met for exposures up to 48.7 Mwd/kg, and have committed to not exceed that limit without analyses which use methodologies acceptable to the NRC. Thus, we conclude that the proposed changes will not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

Criterion 3

See Criterion 1, above.

Lastly, we note that the Commission has provided guidance concerning the determination of significant hazards by providing certain examples (48 FR 14870) of amendments considered not likely to involve significant hazards consideration. The sixth of these examples refers to changes which may result in some increase to the probability of occurrence or consequences of a previously analyzed accident or may reduce in some way a safety margin, but the results of which are clearly within limits established as acceptable. Because these proposed changes involve extension of a limit contained in the T/Ss, they may be perceived as involving a reduction in safety margin; however, for reasons previously presented, we do not believe that any reductions would be significant.

Attachment 4 to AEP:NRC:1018

ANF Evaluation (Proprietary) of Mechanical Design Considerations for Peak Pellet Exposures Up to 48.7 Mwd/kg

FEB 1 1 1987

ADVANCED NUCLEAR FUELS CORPORATION

... -CRN RAPIDS ROAD PO BOX 130 RICHLAND ...
509 375-8100 TELEX 15-2878

February 10, 1987
HGS-87-055(P)

Indiana & Michigan Electric Company
c/o Richard B. Bennett
Engineer, Nuclear Materials & Fuel Mgmt.
American Electric Power Service Corp.
One Riverside Plaza, 20th Floor
Columbus, OH. 43216-6631

Dear Rick:

Subject: <u>D. C. Cook 1 - Peak Pellet Burnup Extension</u>

Attached is a summary report of the D. C. Cook Unit 1 peak pellet burnup extension analysis. This review was conducted to provide an increase in the peak pellet burnup limit from 48,000 to 48,700 MWd/MTU. The peak assembly burnup remains unchanged at 41,000 MWd/MTU. The extension of the peak pellet exposure will not result in the violation of any design criteria.

Advanced Nuclear Fuels Corporation considers information contained in the enclosed technical report to be proprietary. Also enclosed is a non-proprietary version of the report. The Affidavit enclosed provides the necessary information to allow the withholding of the proprietary version from public disclosure as required by 10 CFR 2.790(b).

Very truly yours,

NL Durner for

H. G. Shaw
Contract Administrator

sh

Attachment

xc: M. P. Alexich
J. M. Cleveland
D. H. Malin
V. Vanderburg

DC Cook Unit 1 - Peak Pellet Burnup Extension

Background:

The last reload of ANF (formerly ENC) fuel supplied for the DC Cook Unit 1 reactor is currently in its last cycle of operation. A burnup extension analysis had been performed for this fuel in 1984 in order to support burnup levels of 41.0, 43.7, and 48.0 GWD/MtU respectively for peak assembly, peak rod, and peak pellet. Reactor operating conditions since that time have resulted in higher axial peaking than originally projected. Consequently, the peak pellet burnup is now expected to approach a level of 48.5 GWD/MtU. The peak rod and peak assembly burnup levels are not affected. A review of the original analyses supporting the burnup extension has been conducted in order to determine the consequences of an increase in peak pellet exposure. The review considered an additional increase in peak pellet exposure to 48.7 GWD/MtU to provide margin for a potential end of cycle coastdown.

Summary of Burnup Extension Analysis Review:

The original burnup extension analysis, reported in XN-NF-84-25, Rev. 0 (Reference 1), addressed the following aspects of design: (1) Steady State Stress, (2) Steady State Strain, (3) Cladding Corrosion and Hydrogen Absorption, (4) Transient Stress and Strain and Cladding Fatigue, (5) Cladding Creep Collapse, (6) Fuel Rod Internal Gas Pressure, (7) Fuel Rod Growth, (8) Spacer Spring Force, and (9) Fuel Assembly Growth. Of these, only Steady State Strain, Corrosion and Hydrogen Absorption, and Fuel Rod Internal Pressure are significantly affected by the axial profile of the fuel rod. The remainder of the items are essentially independent of the peak pellet exposure. The results reported in XN-NF-84-25, Rev. 0 remain valid for these items.

The power history used for the original burnup extension analysis was based on a conservative best-estimate of the maximum discharge exposure rod, assuming full power operation. In reality the operation of the reactor has been limited to 90 percent of full power. Therefore, the original power history projection represents a bounding case for this fuel.

The revised analysis shows that clad strain, corrosion and hydrogen absorption remain within the design limits, and the fuel rod pressure remains below system pressure.

Steady State Strain, Cladding Corrosion and Hydrogen Absorption:

The maximum cladding strain, corrosion and hydrogen absorption were determined to occur at the peak axial region in the original burnup extension analysis. Review of this analysis showed the results from the previous analysis to have been taken for a peak pellet exposure of 48.3 GWD/MtU. Because of the substantial margin for these design criteria a simple extrapolation was used to project the conditions for a peak pellet exposure of 48.7 GWD/MtU. Extrapolating the results of the original analysis and including an uncertainty of five percent yields the following results:

	Projected	Criteria
Total Positive Strain, (%)	[none	1.0]
Maximum Positive Strain Increase, (%)	[0.31	1.0]
Cladding Corrosion, (inch)	[0.00073	0.002]
Hydrogen Absorption, (ppm)	[85.	300.]

Therefore, the fuel will remain well within the criteria for these items.

Fuel Rod Internal Pressure:

A new RODEX2 (Reference 2) analysis was performed using the approved methodology for internal gas pressure determination and the bounding power history. The axial peaking factor from the original extension analysis was increased by 2% at the maximum axial region in order to bound the 1.5% increase in burnup from 48.0 to 48.7 GWD/MtU. The results of this analysis showed a peak internal pressure of [1825] psia over the design life of the fuel. This value is well within the criteria limit of the 2250 psia reactor operating pressure as given in XN-NF-84-25, Rev. 0.

Conclusion:

Review of the analysis for the ANF fuel supplied to the DC Cook Unit 1 reactor has shown the fuel capable of meeting all design criteria at a peak pellet exposure of 48.7 GWD/MtU. The results presented in the extended burnup report XN-NF-84-25, Rev. 0 with the addition of the results presented in this letter remain valid for the fuel.

Ref: (1) XN-NF-84-25, Revision 0, Mechanical Design Report Supplement for DC Cook Unit 1 Extended Burnup Fuel Assemblies, April 1984.

(2) XN-NF-81-58 (P)(A), Revision 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, March 1984.

A F F I D A V I T

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

I, H. E. Williamson being duly sworn, hereby say and depose:

1. I am Manager, Licensing and Safety Engineering, for Advanced Nuclear Fuels Corporation ("ANF"), and as such I am authorized to execute this Affidavit.

2. I am familiar with ANF's detailed document control system and policies which govern the protection and control of information.

3. I am familiar with the Letter HGS-87-55(P) entitled "DC Cook Unit 1 Peak Pellet Burnup Extension" referred to as "Document." Information contained in this Document has been classified by ANF as proprietary in accordance with the control system and policies established by ANF for the control and protection of information.

4. The document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by ANF and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in the Document as proprietary and confidential.

5. The Document has been made available to the U.S. Nuclear Regulatory Commission in confidence, with the request that the information contained in the Document will not be disclosed or divulged.

6. The Document contains information which is vital to a competitive advantage of ANF and would be helpful to competitors of ANF when competing with ANF.

7. The information contained in the Document is considered to be proprietary by ANF because it reveals certain distinguishing aspects of PWR Fuel Design methodology which secure competitive advantage to ANF for fuel design optimization and marketability, and includes information utilized by ANF in its business which affords ANF an opportunity to obtain a competitive advantage over its competitors who do not or may not know or use the information contained in the Document.

8. The disclosure of the proprietary information contained in the Document to a competitor would permit the competitor to reduce its expenditure of money and manpower and to improve its competitive position by giving it extremely valuable insights into PWR Fuel Design methodology and would result in substantial harm to the competitive position of ANF.

9. The Document contains proprietary information which is held in confidence by ANF and is not available in public sources.

10. In accordance with ANF's policies governing the protection and control of information, proprietary information contained in the Document has been made available, on a limited basis, to others outside ANF only as required and under suitable agreement providing for non-disclosure and limited use of the information.

11. ANF policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

12. This Document provides information which reveals PWR Fuel Design methodology developed by ANF over the past several years. ANF has invested thousands of dollars and several man-months of effort in developing the PWR Fuel Design methodology revealed in the Document. Assuming a competitor had available the same background data and incentives as ANF, the competitor might, at a minimum, develop the information for the same expenditure of manpower and money as ANF.

THAT the statements made hereinabove are, to the best of my knowledge, information, and belief, truthful and complete.

FURTHER AFFIANT SAYETH NOT.

H. E. Williamson

SWORN TO AND SUBSCRIBED

before me this 14th day of

February, 1988.

[illegible]

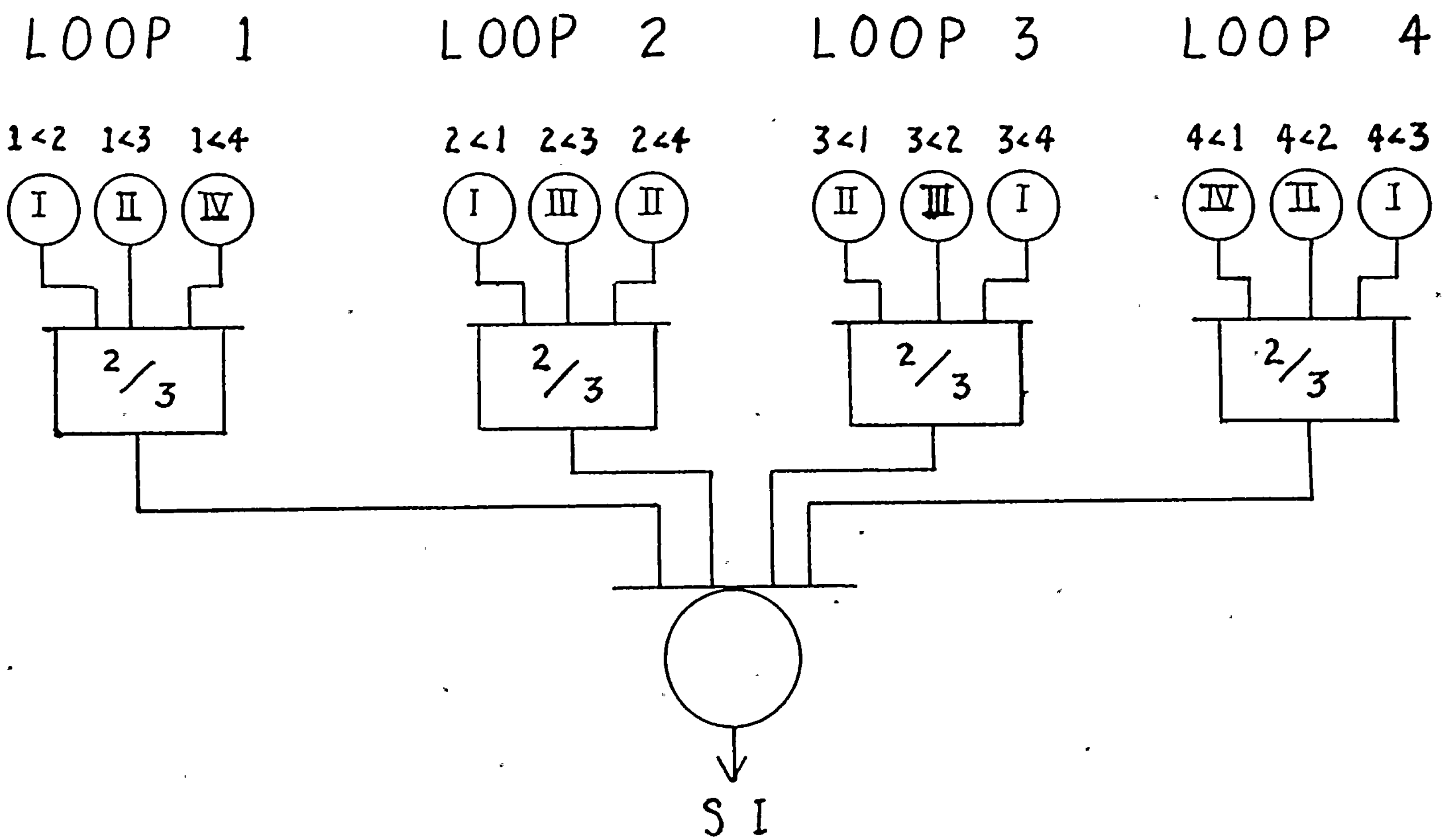
NOTARY PUBLIC

Attachment 16 to AEP:NRC:0916W

EXPLANATION OF STEAMLINE DIFFERENTIAL PRESSURE

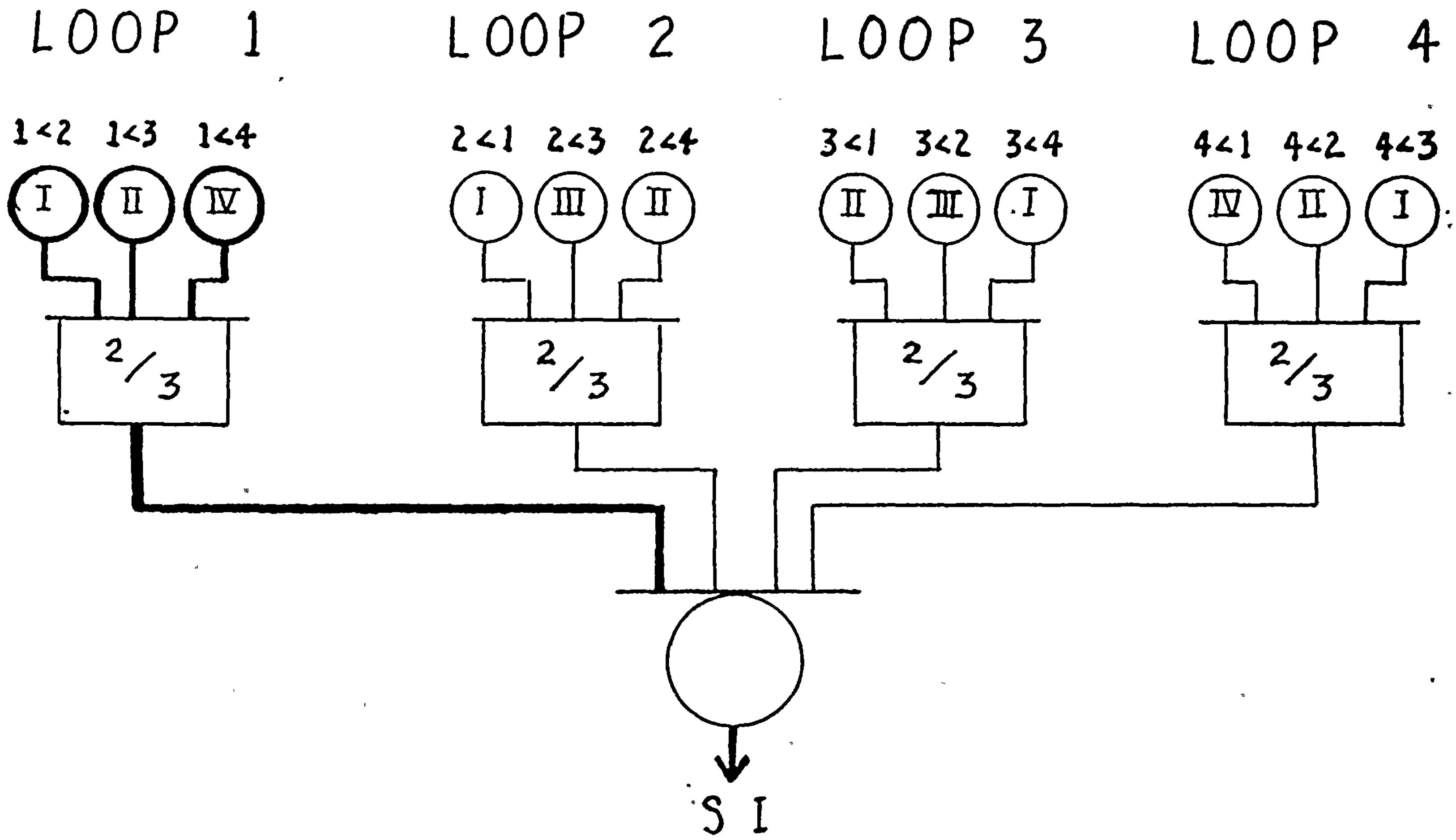
ENGINEERED SAFETY FEATURE

ACTUATION INSTRUMENTATION LOGIC



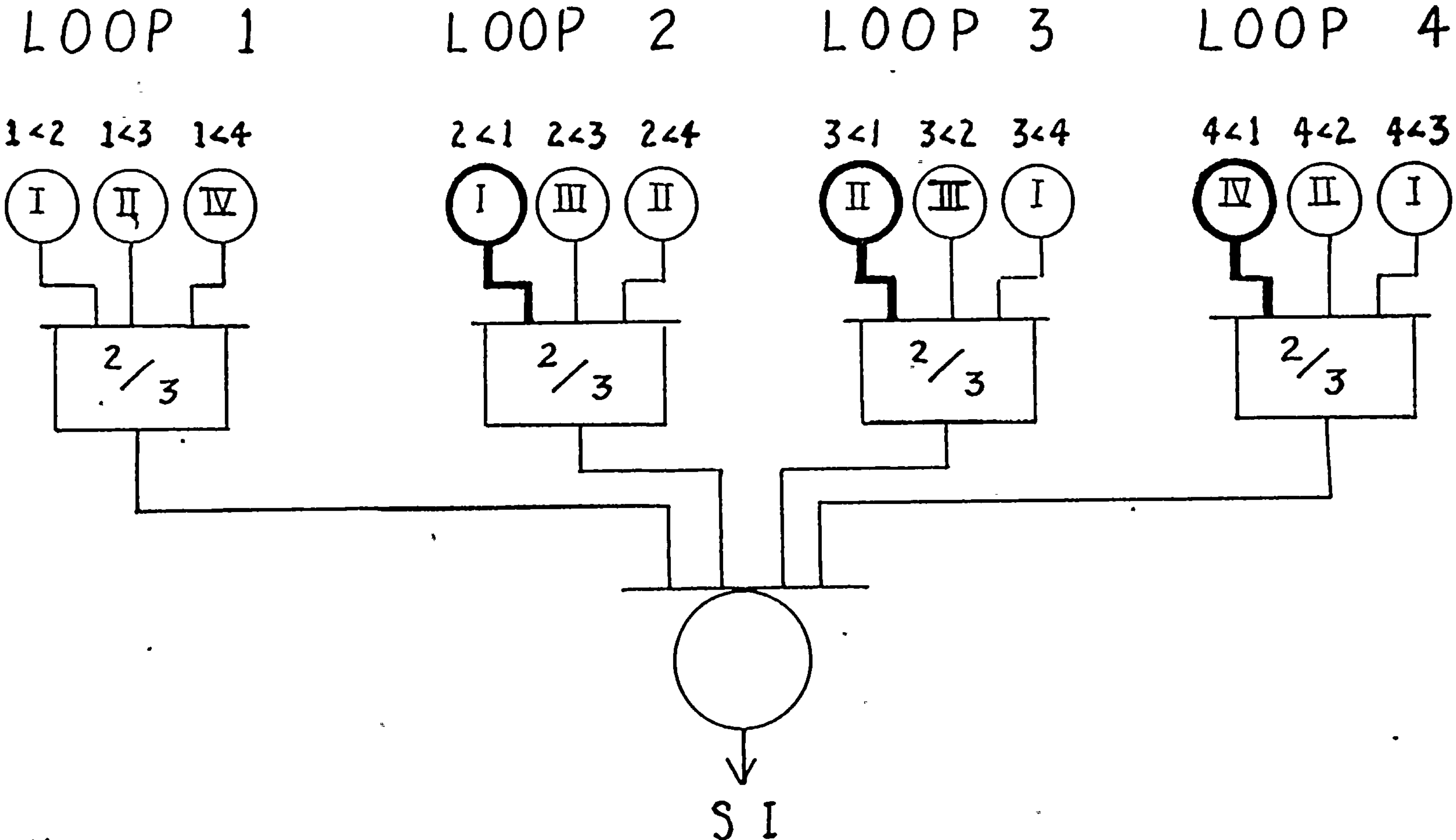
This figure represents the Steamline Differential Pressure logic. The small circles represents individual bistable signals and the enclosed roman numerals indicate the protection channels from which the signal is derived. All bistables are shown in the untripped condition.

FIG. 1



This figure represents the conditions which would occur if the currently approved Technical Specifications are interpreted to mean bistables for the isolated loop should be placed in the tripped condition. In this example, Loop 1 is the isolated loop and the Loop 1 bistables are shown tripped. As soon as the second bistable is tripped, the logic for a safety injection is satisfied and a SI will occur.

FIG. 2



This figure represents the conditions established by the correct interpretation of the Technical Specifications for three loop operation. This interpretation is clarified in the proposed #### footnote of Table 3.3-3 which was approved for Unit 2. Again, Loop 1 is assumed to be the isolated loop. As indicated, only the operating loop bistable which compares the operating loop's pressure relative to the isolated loop's pressure is placed in the tripped condition. This action reduces the Steamline Differential Pressure SI logic to a one per steamline in any operating loop. This is what the Technical Specifications require.

Tripping the indicated bistables does not diminish the protection available for a steamline break. Should the break occur in one of the operable steamlines, the protective action will occur as soon as the pressure of the affected steam generator falls sufficiently below that of either of the two remaining operable steam generators. If the break occurred in the isolated loop, the normal protective logic would be present to provide protection. There has been no compromise of the isolated loop's logic.

Attachment 17 to AEP:NRC:0916W

COPY OF LETTER DATED JULY 9, 1984 FROM E. P. RAHE (WESTINGHOUSE) TO

D. EISENHUT (NRC) (NS-TA-84-003)

COPY OF INDIANA AND MICHIGAN ELECTRIC COMPANY

LETTER AEP:NRC:0895

Westinghouse
Electric Corporation

Water Reactor
Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

July 9, 1984

NS-TA-84-003

Mr. D. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
2920 Norfolk Avenue
Washington, D.C. 20555

Dear Mr. Eisenhut:

NUMBER OF OPERATING REACTOR COOLANT PUMPS IN MODE 3

This letter formalizes the material presented on June 15, 1984, with respect to the consistency between the Technical Specifications and the safety analysis for the number of operating reactor coolant pumps in Mode 3. This meeting was held at the request of the NRC staff in order to discuss the Westinghouse determination of a potential unreviewed safety question for three and four loop plants for this issue. Enclosed are ten (10) proprietary copies of the slides and ten (10) non-proprietary copies. Also enclosed are one (1) copy of Application for Withholding, AW-84-63 (non-proprietary) and one (1) copy of Affidavit (non-proprietary).

As part of an informal review of a utility's Tech Specs by the NRC Reactor Systems Branch, the staff asked what the safety analysis assumptions were concerning the number of operating reactor coolant pumps, particularly at or near zero power. Although the question was never formally asked, Westinghouse reviewed the analysis assumptions with respect to the Tech Specs.

The requirement for operating reactor coolant pumps under these conditions is contained in Specification 3.4.1.2 of the Standard Tech Specs. In non-Standard Tech Specs, the requirement is contained in Specification 3.1. These Specs state that when the plant is subcritical by the shutdown margin between 350°F (RHR cut-in) and 547°F or 557°F (no-load conditions), there must be two loops operable, but only one loop has to be actually operating.

However, the safety analysis in the FSARs assumes that either two or all of the reactor coolant pumps are operating, not just one. (At the staff's request, the assumptions made concerning the number of operating pumps have been noted for those plants within Westinghouse scope in the attachment). The accidents which are limiting at zero power are steamline break, rod ejection, and bank withdrawal from subcritical. Westinghouse has reviewed these accidents under the reduced flow conditions of one pump. For the rod ejection and steamline break events, Westinghouse has determined that the inconsistency between the safety analysis

and the Tech Spec will not impact the conclusions presented in the FSAR. For the bank withdrawal from subcritical event, Westinghouse has performed calculations which show that the DNB design basis may not be met when only one pump is in operation. Thus, the margin of safety as defined in the basis of the Tech Specs is reduced.

Westinghouse has also performed calculations for one pump operation assuming more realistic, but still conservative, reactivity insertion rates. The results of these calculations show that the DNB design basis is met. Other assumptions and models used in these analyses are identical to the FSAR methods of analysis for this event. Thus, Westinghouse feels that no significant safety hazard exists.

Westinghouse is currently considering long term analytical solutions to this issue which will show that the DNB design basis can be met when only one reactor coolant pump is in operation so that the Tech Specs will not need to be changed. However, in the short term, Westinghouse recommends that the plants be operated with the same number of reactor coolant pumps in operation as was assumed in the analysis. Note that this is not a realistic requirement when the plant is cooling down prior to going into Mode 4 (RHR operation), particularly for those plants for which the analysis assumes all pumps in operation. Thus, an alternative to having more than one pump in operation is to prevent rod withdrawal. This will preclude the accident from taking place. Although physical prevention of withdrawal will accomplish this, administrative procedures may be preferable. The ability to cock the rods partway out of the core during Mode 3 provides desired operating flexibility. Furthermore, there is no mechanism by which the control rods can be automatically withdrawn in Mode 3 due to a control system error. Increased operator awareness during this time and adherence to procedures will also prevent the accident from occurring.

Finally, while Westinghouse feels that it is appropriate to consider bank withdrawal when in Mode 3, Westinghouse does not intend to address this event in other modes of operation (Standard Tech Spec Modes 4 and 5). Bank withdrawal from subcritical is a valid scenario when going from Mode 3 to Mode 2. However, consideration of bank withdrawal in Modes 4 and 5 is unrealistic and it is questionable as to whether it is applicable or if it is a Condition II event. Again, increased operator awareness must be considered when evaluating the appropriateness of the event.

Correspondence with respect to the Westinghouse affidavit or application for withholding should reference AW-84-63, and should be addressed to Mr. R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, P.O. Box 355, Pittsburgh, Pennsylvania 15230. Other correspondence or questions should be directed to Mr. J. L. Little, Manager, Operating Plant Licensing Support, 412/374-5054.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

E. P. Rahe, Jr.
Nuclear Safety Department

M. P. Osborne/ds

Enclosures

STS PLANTS

OPERATING	NON-OPERATING
D. C. Cook 1	Seabrook 1 & 2
Salem 1 & 2*	Catawba 1 & 2
Beaver Valley 1*	Byron/Braidwood
Diablo Canyon 1 & 2	Beaver Valley 2
McGuire 1 & 2	Vogtle 1 & 2
Summer*	Millstone 3
Farley 1 & 2*	Comanche Peak 1 & 2
Sequoyah 1 & 2*	Watts Bar 1 & 2*
Trojan*	South Texas 1 & 2
	Shearon Harris 1 & 2
	Marble Hill 1 & 2

NON-STS PLANTS

Turkey Point 3 & 4*
Zion 1 & 2*
Indian Point 2 & 3*

(*) Assumes all pumps operating

PLANTS OUTSIDE W SCOPE

D. C. Cook 2	Yankee Rowe
Robinson 2	Surry 1 & 2
Haddam Neck	North Anna 1 & 2

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

July 30, 1984
AEP:NRC:0895

Donald C. Cook Nuclear Plant Unit Nos. 1 and 2
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
NUMBER OF REACTOR COOLANT PUMPS OPERATIONAL IN MODE 3

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

By letter dated June 6, 1984, Indiana & Michigan Electric Company (IMECo) was notified by Westinghouse Electric Corporation (W) that several Final Safety Analysis Report (FSAR) analyses performed at Hot Zero Power (HZP) assumed the operation of two (2) Reactor Coolant Pumps (RCPs). The limiting analyses at HZP, i.e., steam line break, rod ejection, and bank withdrawal from subcritical conditions, are assumed to bound postulated Operational Mode 3 accidents and transients. The Donald C. Cook Nuclear Plant Unit Nos. 1 and 2 Appendix "A" Technical Specification (T/S) 3.4.1.2, however, requires that only one (1) RCP be operating during Operational Mode 3, and that at least one (1) additional RCP be available to meet single failure criteria.

The attachment to this letter contains a copy of the notification which we received from W. As noted in this letter, W has determined that the inconsistency between the FSAR and the T/S will not impact the FSAR conclusions for the steam line break accident and the rod ejection transient. For the bank withdrawal from subcritical conditions transient, W calculations indicate that the departure from nucleate boiling (DNB) design basis may not be met when only one (1) RCP is running. On a best estimate basis, however, W believes that ". . . the DNB design basis can be met. The FSAR licensing basis analysis includes conservatisms (such as high reactivity insertions rates) which when removed, show that [departure from nucleate boiling ratio] DNBR is above the limit value. Thus, no significant safety hazard exists. . . ."

We are currently preparing a proposed amendment to the T/S to deal with this situation. In the interim period until the modified T/S is approved by your staff, we have instituted a temporary procedural change to ensure that plant operations are consistent with the FSAR analysis assumptions. That instruction requires that we operate with at least two (2) reactor coolant pumps while in Mode 3 unless the reactor trip breakers are disconnected.

We are notifying you consistent with 10CFR50.36. This matter was discussed with your staff upon notification from W.

This document has been prepared following Corporate procedures which incorporate a reasonable set of controls to ensure its accuracy and completeness prior to signature by the undersigned.

M. P. Alexich
Vice President

MPA/dam
Attachment

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Charnoff
E. R. Swanson - NRC Resident Inspector, Bridgman

ATTACHMENT TO AEP:NRC:0895
WESTINGHOUSE LETTER REGARDING NUMBER OF
REACTOR COOLANT PUMPS IN OPERATIONAL MODE 3
DONALD C. COOK NUCLEAR PLANT UNIT NOS. 1 AND 2

Westinghouse
Electric Corporation

Water Reactor
Divisions

Nuclear Services
Integration Division

Box 2728
Pittsburgh Pennsylvania 15230-2728

JUN 12 1984
COOK PLANT
MANAGERIAL

June 6, 1984
AEP-84-612

Mr. W. G. Smith, Plant Manager
D. C. Cook Nuclear Plant
Indiana and Michigan Power Company
P. O. Box 458
Bridgman, Michigan 49106

Dear Mr. Smith:

American Electric Power Service Corporation
D. C. Cook Unit 1
CONSISTENCY BETWEEN SAFETY ANALYSIS AND TECHNICAL SPECIFICATIONS
CONCERNING NUMBER OF REACTOR COOLANT PUMPS IN OPERATION

This letter is to notify you of a potential unreviewed safety question concerning the consistency between the safety analysis and the Technical Specifications. According to 10CFR50.36, the assumptions in the safety analysis and the plant Tech Specs must be consistent. This ensures that the plant is operated in a manner such that it is bounded by the FSAR accident analysis.

As part of an informal review of a utility's Tech Specs in the NRC Reactor Systems Branch, the staff asked what the safety analysis assumptions were concerning the number of operating reactor coolant pumps, particularly at or near zero power. This information is stated in the FSAR for the zero power accidents. Although the question was never formally asked, Westinghouse reviewed the analysis assumption with respect to the Tech Specs.

The issue in question concerns the number of operating reactor coolant pumps when in Mode 3, which is defined in the Tech Specs as between 350°F and the no-load temperature (either 547 or 557°F). The reactor is also subcritical as required by the Shutdown Margin Spec, Standard Tech Spec 3.1.1.1. The STS Spec number (which should correspond to your Spec number) which contains the requirement for the number of operating loops is Spec 3.4.1.2. This Tech Spec states that in Mode 3, there must be two loops operable (which means that the reactor coolant pump must be operable), but only one loop must be actually operating.

4.1.4.1

However, the safety analysis in the FSAR assumes that either two or all of the reactor coolant pumps are actually operating, not just one. In the FSAR, analyses performed at Hot Zero Power (HZP) are assumed to bound Mode 3 operation. The accidents which are limiting at HZP are steamline break, rod

ejection and bank withdrawal from subcritical. Westinghouse has reviewed these accidents under the reduced flow conditions of one pump. For the rod ejection and steamline break events, Westinghouse has determined that the inconsistency between the safety analysis and the Tech Spec will not impact the conclusions presented in the FSAR. However, for the bank withdrawal from subcritical accident, Westinghouse has performed calculations which show that the DNB design basis for this Condition II event may not be met when only one pump is in operation. Thus, the margin for safety as defined in the basis for the Tech Specs is reduced and this may be an unreviewed safety question according to 10CFR50.59.

Note that on a best estimate basis, the DNB design basis can be met. The FSAR licensing basis analysis includes conservatisms (such as high reactivity insertions rates) which when removed, show that the DNBR is above the limit value. Thus, no significant safety hazard exists.

Westinghouse recommends that you review your FSAR analysis for the bank withdrawal from subcritical event for consistency with your Tech Specs. Furthermore, Westinghouse recommends that you require the number of operating pumps in Mode 3 to be consistent with the analysis. Alternatively, you should ensure that rod withdrawal will not occur when in Mode 3 if the requirement for pump operation cannot be met in Mode 3. This will ensure that the safety analysis is consistent with plant operation.

If you have any questions, please contact me.

Very truly yours,

W. J. Johnson

W. J. Johnson, Manager
Projects Department
Central Area

HT/387L

cc: M. P. Alexich
W. G. Smith
J. Waleko W

Attachment 19 to AEP:NRC:0916W

REVIEW OF THE PROPOSED POWER DISTRIBUTION TECHNICAL SPECIFICATION

SIMPLIFICATIONS PERFORMED BY

WESTINGHOUSE ELECTRIC CORPORATION

LETTER FROM WESTINGHOUSE ELECTRIC CORPORATION

SUPPORTING A BURNUP INDEPENDENT F_Q FOR

WESTINGHOUSE FUEL TO AT LEAST 60 MWD/KG PEAK PELLET BURNUP

Westinghouse Electric Corporation

Nuclear Fuel Divisions

Box 3912
Pittsburgh Pennsylvania 15230-3912

87AE*-G-0010
January 23, 1987

W-AEP/0324

KEYWORDS:
AEP
TECH-SPEC

Indiana and Michigan Electric Company
c/o Eric G. Lewis
Engineer, Nuclear Materials and Fuel Management
American Electric Power Service Corporation
One Riverside Plaza, 20th Floor
Columbus, OH 43215

Dear Mr. Lewis:

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D. C. COOK UNIT 1
TECHNICAL SPECIFICATION SIMPLIFICATION

As requested by American Electric Power Service Corporation (AEPSC) in AEP-W/0151, Westinghouse has reviewed your proposed simplification of Sections 3.2.2 and 3.2.6 of the D. C. Cook Unit 1 Technical Specifications. The changes include removal of burnup dependence in the heat flux hot channel factor limit and allowable power level for EXXON fuel.

Westinghouse has found the proposed changes to be consistent with the design basis for D. C. Cook Unit 1 and the Westinghouse reload methodology.

Very truly yours,

N. E. Campbell

N. E. Campbell
Project Engineer, NFD Projects

NEC:mld

cc: M. P. Alexich
J. M. Cleveland
D. H. Malin
V. D. Vanderburg

MAR 7 1986

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Fuel Division

Box 3912
Pittsburgh Pennsylvania 15230-3912

March 3, 1986

86AE*-G-0020

W-AEP/0244

Keywords: AEP
Tech-Spec

Indiana and Michigan Electric Co.
c/o Joseph L. Bell
Engineer, Nuclear Materials and Fuel Management
American Electric Power Service Corp.
One Riverside Plaza, 20th Floor
Columbus, OH 43215

Dear Mr. Bell:

AMERICAN ELECTRIC POWER SERVICE CORPORATION
D.C. COOK UNIT 1
AEP TECH SPEC CHANGE

Please find attached pages of the D.C. Cook Unit 1 Tech. Spec. which have been marked up to reflect the extension of the FQ exposure dependent limit to 60 MWD/Kg. This was informally given to you at our meeting on February 28, 1986.

As per your request, the current Tech. Spec format has been maintained with Ep(Z) = 1.0, T(El) = 1.0, and FQ (El) = 2.10 for a peak pellet exposure extending from 0.0 to 60.0 MWD/Kg.

If you have any questions, please call me.

Very truly yours,

J.C. Miller
Project Engineer
NFD Fuel Projects

/kph

cc: M.P. Alexich
J.M. Cleveland
D.H. Malin - w/enc.
V.D. Vanderburg
W.L. Zimmermann

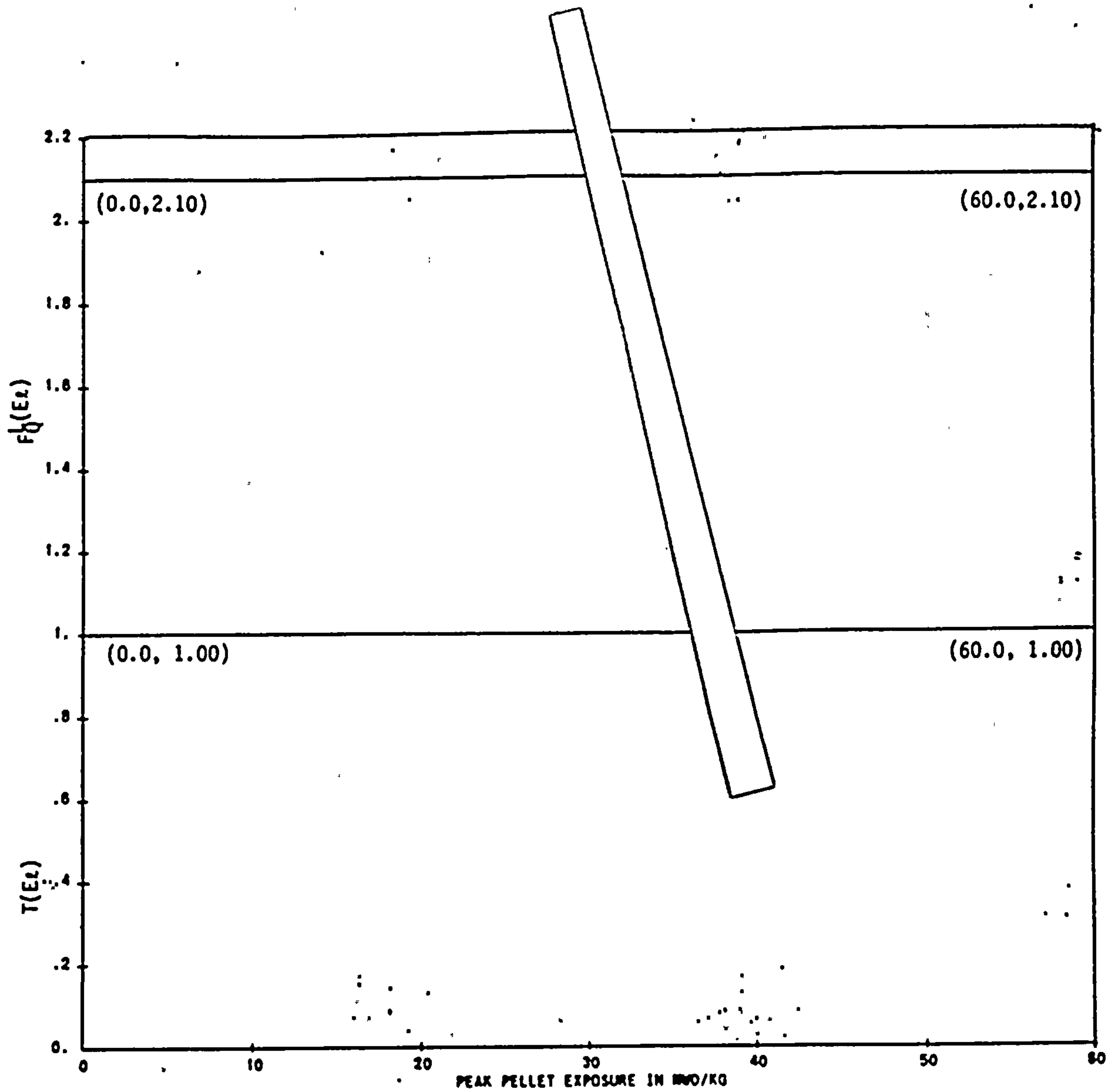


FIGURE 3.2-5

Exposure Dependent F_Q Limit, $F_Q^L(E_l)$, and Normalized Limit $T(E_l)$ as a Function of Peak Pellet Burnup for Westinghouse Fuel

Attachment 20 to AEP:NRC:0916W

LIST OF RETRANSMITTED PROPRIETARY DOCUMENTS

WHICH ARE REQUESTED BE WITHHELD

List of Resubmitted Proprietary Documents

	Proprietary Document	AEP:NRC:0916W Attachment Number	Previous Submittal
1.	AEP-D.C. Cook Unit 1 RdF RTD Installation Safety Evaluation August 6, 1985	4	Indiana & Michigan Letter AEP:NRC 0942D, dated August 13, 1985
2.	Safety Evaluation for Operation Between the Time RTD Cross Calibration Data is Obtained and Calibration is Updated	8	Indiana & Mighigan Letter AEP:NRC:0942D dated August 13, 1985
3.	XN-NF-85-115(P) Rev. 2 D. C. Cook Unit 1 Limiting Break K(Z) LOCA/ECCS Analysis	15	Exxon (Now Advanced Nuclear Fuels) Letter GNW:001:87, dated January 15, 1987
4.	Advanced Nuclear Fuels Evaluation of Mechanical Design Considerations for Peak Pellet Exposures up to 48.7 MWd/kg	15	Indiana & Michigan Letter AEP:NRC:1018 dated February 20, 1987
5.	American Electric Power D. C. Cook Unit 2 RdF RTD Installation Safety Evaluation	18	Indiana & Michigan Letter AEP:NRC:0916I dated March 14, 1986

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