ATTACHMENT B

UNIT 2 PROPOSED SPECIFICATIONS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.1 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

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

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

MAR 01 1985

ATTACHMENT D

UNIT 3 PROPOSED SPECIFICATIONS

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.1 weight percent U-235.

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5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

b. For a pressure of 2500 psia, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

MAR 01 1985

ATTACHMENT E

ANALYSIS SUMMARY

LEAnderson
6/4/85

JUSTIFICATION FOR STORAGE OF 4.1 W/O ENRICHED FUEL IN SAN ONOFRE 2/3 FUEL STORAGE RACKS

1. PURPOSE

Criticality calculations were performed to verify that the SONGS 2/3 spent and new fuel storage racks and the fuel transfer carrier can acceptably store 4.1% enriched fuel from a criticality standpoint. The original design was performed for 3.7% enriched fuel and contained sufficient margin to keep the worst k-effective in the spent fuel storage rack below 0.95 and the worst k-effective of the new fuel storage racks below 0.98. The original design calculations are documented in References 1 and 2.

2. METHODOLOGY

The method for calculating the criticality of various fuel storage and fuel handling configurations is comprised of the following computer programs:

1. KENO-IV/S
2. NITAWL-S
3. EPRICELL-2
4. DANCOFF

The KENO-IV/S and NITAWL-S programs were obtained from the Technical Data Management Center at Oak Ridge National Laboratory as part of the SCALE-2 Computer Code System (Reference 3). The EPRICELL-2 and DANCOFF were obtained from the Electric Power Research Institute as part of the ARMP code package (Reference 4).

KENO-IV/S is a multi-group Monte Carlo criticality program. It utilizes a very simple geometry input that is capable of modeling three-dimensional systems exactly. The principal result calculated by KENO-IV/S is the system k-effective, the estimate of criticality.

NITAWL-S performs the resonance shielding calculations for these nuclides which have resonance parameters included in their basic cross section data. NITAWL-S uses the Nordheim Integral treatment for the resonance calculations. The 27-group neutron cross section library furnished with the SCALE package was used for these analyses for input to both NITAWL-S and KENO-IV/S.

EPRICELL-2 is an EPRI computer program for fuel pin cell calculations. In these analyses it was used to generate fuel material concentrations for the 4.1% enriched fuel and the DANCOFF factors for input to NITAWL-S.

The DANCOFF program calculates the DANCOFF factor based on Lauer's method and was used to independently check the values calculated by EPRICELL-2 and to calculate values at water densities below the limit at which EPRICELL ceases to function.

The basic sequence of calculations is as follows:

1. Generation of Dancoff factors at various water densities and calculation of fuel nuclide concentrations using EPRICELL-2 and DANCOFF.

2. Generation of 27 group neutron libraries for use in KENO-IV/S with the NITAWL-S program calculating resonance effects.

3. KENO-IV/S calculation of the reference configuration using nominal conditions.

4. KENO-IV/S calculations with normal and abnormal variations of system conditions (i.e. pitch, water density, fuel handling accident, etc.).

5. The reference case k-effective is combined with the calculation-to-measured bias from Reference 5 and all the normal and abnormal k-effective variations to estimate the worst system k-effective.

The system worst k-effective is then compared to the design limits specified in NRC Regulatory Guide 1.13(6) or the NRC standard review plan (7), whichever is applicable. These limits are 0.95 for spent fuel storage racks and fuel handling systems and 0.98 for new fuel storage racks.

3. RESULTS

Spent Fuel Rack

Reference Case k-effective	0.90111
KENO-IV Calculation Uncertainty $\Delta k (2\sigma)$	0.00266
Calculation-to-Measured Bias Δk (from DC-1859)	0.01332
Worst Normal Conditions Δk Minimum Rack Pitch & Eccentric Fuel Load	0.01503
Worst Abnormal Temperature Δk (39°F)	0.00192
Dropped Fuel Assembly Accident Δk	0.00214
Waterlogged fuel pins (1% failed) Δk	0.00008
Worst k-effective	0.93626

New Fuel Rack

Reference Case k-effective	0.38218
KENO-IV Calculation Uncertainty Δk(2 σ)	0.00208
Calculation-to-measured Bias Δk (from DC-1859)	0.01332
Normal Humidity Range Δk	0.01301
Accident Δk (Worst of flooding or fuel handling accident)	max (0.46567,0.08527)
Worst k-effective	0.87626

Fuel Transfer Carrier

Worst Temperature k-effective	0.89226
KENO-IV Calculation Uncertainty Δk(2 σ)	0.00232
Calculation-to-measured Bias Δk (from DC-1859)	0.01332
Worst k-effective	0.90790

4. REFERENCES

1. NES 81A0529, Rev. O, "Nuclear Design Analysis Report for the San Onofre Nuclear Generating Station Units 2 and 3 High Density Fuel Storage Racks", December 16, 1977.

2. NES 81A0534, Rev. O, "Nuclear Design Analysis Report for the New Fuel Storage Racks for the San Onofre Nuclear Generating Station Units 2 and 3", October 6, 1978.

3. NUREG/CR-0200, "SCALE-2 A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Technical Data Management Center, Oak Ridge National Laboratory.

4. "Advanced Recycle Methodology Program System Documentation", Electric Power Research Institute, CCM-3, January 1976.

5. "KENO-IV Benchmarking Calculations for SONGS 2/3 Criticality Analyses", C. W. Gabel, May 1985, DC-1859.

6. NRC Regulatory Guide 1.13 Revision 1, December 1975.

7. NRC Standard Review Plan, NUREG-0800, Section 9.1.1, Pages 9.1.1-9.1.4, "New Fuel Storage".

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-200

This is a request to revise Technical Specification (TS) 3/4.1.2.1, "Boration Systems - Flow Path - Shutdown," TS 3/4.1.2.2, "Boration System - Flow Paths - Operating," TS 3/4.1.2.6, "Boric Acid Makeup Pumps - Operating," TS 3/4.1.2.7, "Borated Water Source - Shutdown," TS 3/4.1.2.8, "Borated Water Source - Operating," TS 3/4.5.1, "Safety Injection Tanks," TS 3/4.5.4, "Refueling Water Storage Tank," and Associated Bases.

Existing Specifications:

Unit 2: See Attachment A
Unit 3: See Attachment C

Proposed Specifications:

Unit 2: See Attachment B
Unit 3: See Attachment D

Description

The proposed change will revise Technical Specifications 3/4.1.2.1, "Boration Systems - Flow Paths - Shutdown," TS 3/4.1.2.2, "Boration System - Flow Paths - Operating," TS 3/4.1.2.6, "Boric Acid Makeup Pumps - Operating," TS 3/4.1.2.7, "Borated Water Sources - Shutdown," TS 3/4.1.2.8, "Borated Water Source - Operating," TS 3/4.5.1, "Safety Injection Tanks," TS 3/4.5.4, "Refueling Water Storage Tank," and Associated Bases.

These specifications define: 1) the volume and concentration of boric acid to be maintained in the refueling water storage tank (RWST) for mitigation of a loss of coolant accident (LOCA) and cooldown of the plant; 2) the concentration and volume of boric acid to be maintained in the safety injection tanks (SIT) for mitigation of a LOCA; 3) the volume and concentration of boric acid to be maintained in the boric acid makeup tanks (BAMU); and 4) the flow paths from the borated water sources to the reactor coolant system required during the various modes of plant operation.

The proposed change will: 1) increase the upper limit on boron concentration for the RWST and SIT's from 2300 to 2500 ppm; 2) reduce the maximum BAMU concentration to allow elimination of heat tracing; and 3) decrease the lower limit on SIT boron concentration for Unit 3, Cycle 2 only from 1720 to 1420 ppm; and, 4) better define boration flow path requirements.

TS 3/4.5.1, "Safety Injection Tanks," and 3/4.5.4, "Refueling Water Storage Tank" specify a maximum boron concentration for the RWST and SIT'S of 2300 ppm. The limit on maximum boron concentration is to ensure that boric acid does not precipitate in the core following a LOCA and hinder core coolability. The proposed change increases the upper limit from 2300 ppm to 2500 ppm.

Specification 3/4.1.2.8, "Borated Water Source - Operating" requires that at least one boric acid makeup tank and associated heat tracing circuit be operable, with the contents of the tank as specified in Figure 3.1-1. In Modes 1 through 4, Figure 3.1-1 requires a minimum of 5450 gallons at a minimum boric acid concentration of 8.5 wt% be maintained. TS 3/4.1.2.8 also requires that the refueling water storage tank be operable containing a minimum of 362,800 gallons borated water with concentration between 1720 and 2300 ppm. TS 3/4.1.2.2 requires a minimum of two of the following three paths: 1) a BAMU tank gravity feed path and associated heat tracing; 2) a BAMU tank path via a boric acid makeup pump and associated heat tracing; or, 3) the gravity feed path from the RWST to the charging pump suctions.

The purpose of the required borated water sources and flow paths to the RCS is to ensure that sufficient borated water is available to maintain the reactor subcritical and provide makeup water to account for reactor coolant system (RCS) shrinkage during cooldown to cold shutdown conditions.

The proposed change will revise TS 3/4.1.2.8 to reduce the concentration range to be maintained in one or both BAMU tanks and to eliminate heat tracing. The range of volumes and concentrations to be maintained in the BAMU tanks is specified in a revised Figure 3.1-1. The revised Figure 3.1-1 will represent the minimum required volume at a given concentration of boric acid to be maintained. The proposed change will allow this volume to be maintained as a combined volume in either or both of the BAMU tanks. The concentration range is approximately 2.25 to 3.5 wt% boric acid. The BAMU tank boric acid requirements depend on the concentration currently maintained in the refueling water storage tank since both are required to provide boration during plant cooldown. Figure 3.1-1 incorporates four curves which represent the minimum boric acid volume required from BAMU tanks for a given RWST concentration.

TS 3/4.1.2.2 "Boration System - Flow Paths - Operating" is revised to reflect the flow paths required from the borated water sources credited in meeting TS 3/4.1.2.8. Currently, 3/4.1.2.2 requires two out of three available flow paths. TS 3/4.1.2.2 also requires heat tracing to be operable and verification at least once per 18 months that the flow paths from the BAMU tanks are capable of delivering a flow of at least 40 gpm to the reactor coolant system (RCS). TS 3/4.1.2.6, "Boric Acid Makeup Pumps - Operating," requires the boric acid makeup pump, in any boration flow path credited in meeting the requirements of TS 3/4.1.2.2, to be operable with an operable emergency power source. The proposed change will require that both the gravity feed and boric acid makeup pump paths from any BAMU tank credited in satisfying the boric acid requirements of 3/4.1.2.8 be operable and that the RWST gravity feed path to the charging pumps be operable. There is one gravity feed path and one boric acid pump path from each of the BAMU tanks. If the combined volume of boric acid required is contained as a combined volume between the two BAMU tanks, then the proposed change will require both gravity feed paths and both boric acid makeup pump paths to be operable. If the boric acid requirements are being satisfied by one BAMU tank, then the proposed change will require only the gravity feed path and the boric acid makeup pump path from that tank to be operable. Since the proposed change always requires a boric acid makeup pump path to be operable, TS 3/4.1.2.6 is revised, for consistency, to reflect that a BAMU pump is always required. The TS 3/4.1.2.6 requirement that operable BAMU pumps be capable of being powered from an operable emergency power source is deleted since it is redundant to the requirements of TS 3/4.8.1.1, "Electrical Power Systems - AC Sources." The proposed change will require the RWST gravity feed path to be operable at all times regardless of the BAMU tank flow paths required. The proposed change removes the requirement for heat tracing from Technical Specification 3/4.1.2.2 and the requirement to verify at least one per 18 months that the flow paths from the BAMU tanks are capable of delivering a flow of at least 40 gpm to the RCS. Both of these requirements relate to the potential for obstruction of the boration flow paths due to precipitation of boric acid at the high concentrations currently required. Limiting the maximum concentration in the BAMU tanks to 3.5 wt% eliminates the need for for heat tracing and flow verification.

TS 3/4.1.2.7, "Borated Water Source - Shutdown" and 3/4.1.2.1, "Boration System - Flow Path - Shutdown" define the borated water sources and flow paths to be maintained during cold shutdown and refueling operations. TS 3/4.1.2.7 requires either one boric acid makeup tank and associated heat tracing to be operable with the tank's contents in accordance with Figure 3.1-1, or the refueling water storage tank with a minimum contained volume of 9,970 gallons with a minimum boron concentration of 1720 ppm. TS 3/4.1.2.1 requires that one boron injection flow path and associated heat tracing circuit be operable from the credited borated water source. The available flow paths are: 1) a BAMU tank gravity feed path to the charging pump suctions; 2) a boric acid makeup pump path to the charging pump suctions; 3) a gravity feed path from the RWST to the charging pump suctions, or: 4) a flow path from the RWST to the RCS via a high pressure safety injection pump. The proposed change revises TS 3/4.1.2.7 to require a minimum of 5,150 gallons to be maintained in a BAMU tank or the RWST with a minimum concentration of 1720 ppm. The proposed change deletes the requirement for heat tracing from TS 3/4.1.2.1 and TS 3/4.1.2.7.

TS 3.4.5.1, "Safety Injection Tanks" requires a minimum concentration of 1720 ppm to be maintained in the SIT tanks. The minimum concentration ensures that the reactor would be sub-critical during reflood following a large break LOCA. The proposed change revises the minimum concentration required in the SIT tanks from 1720 to 1420 ppm for Unit 3, Cycle 2 only.

<u>Safety Analysis</u>

The proposed change described above shall be deemed to involve significant hazards considerations if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with the proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No

The proposed change described above revises borated water source requirements. Borated water sources and flow paths are credited in the analysis of the small break LOCA and are required for both negative reactivity insertion and RCS inventory makeup during plant cooldown.

The small break LOCA relies on flow from a charging pump to augment the flow from the high pressure safety injection pumps. During the small break LOCA, a safety injection signal will start the charging pumps and open the valves in the gravity feed and BAMU pump flow paths from the BAMU tanks and start the BAMU pumps. The proposed change revises the minimum flow paths required to be operable and the volume requirements for the BAMU tanks. The proposed change is more restrictive than the existing TS in that it requires both paths from any credited BAMU tank to be operable at all times, whereas the existing operation could require only one flow path to be operable. The minimum volume requirements for the BAMU tanks ensures that sufficient water is available for injection by the charging pumps during a small break LOCA. Therefore, the proposed change does not affect the probability or consequences of the small break LOCA.

Water from the BAMU tanks and RWST are required for reactivity control and makeup during cooldown. The safe shutdown analysis assumes that the letdown line is not available, in conjunction with a loss of offsite power and a limiting single failure. For this scenario, enough borated water must be available to maintain reactivity control and to makeup for RCS shrinkage.

Assuming that the RWST outlet valve in the gravity feed path to the charging pump suctions fails closed, from the standpoint of RCS shrinkage in this limiting scenario, sufficient water must be available from the BAMU tanks (one or both in combination) to provide makeup to allow for plant cooldown to the point where the plant is depressurized sufficiently to allow injection of water into the RCS from the RWST using the HPSI pumps. From this point on, sufficient water must be available from the RWST to makeup for shrinkage to reach cold shutdown conditions. The proposed volume and flow path requirements will ensure that the plant can be brought to cold shutdown conditions assuming letdown is unavailable, in conjunction with the loss of offsite power, and assuming the limiting single failure.

From the standpoint of reactivity control, the BAMU and RWST concentrations ensure that a minimum of 5.15% shutdown margin is maintained during cooldown to cold shutdown in the above described safe shutdown scenario. Therefore, the proposed change does not adversely affect the safe shutdown analysis.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated.

   Response: No

   The proposed change does not physically alter the configuration of the plant and, therefore, does not create the possibility of a new or different kind of accident from any previously accident evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No

   As described in the response to Question 1 above, the proposed change maintains the analyzed results of the small break LOCA. In addition, the more restrictive requirements on boron flow paths more effectively ensure that the plant can be brought to cold shutdown in the limiting safe shutdown scenario. Therefore, the proposed change does not reduce a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously analyzed accident or may in some way reduce a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP).

The pertinent acceptance criteria for the above described changes are found in SRP Sections 4.3, "Nuclear Design," Section 5.4.7, "Residual Heat Removal System," Section 6.3, "Emergency Core Cooling System," and Section 9.3.4 "Chemical and Volume Control System."

The proposed change would increase the maximum concentration allowed in the SIT and refueling water storage tanks from 2300 ppm to 2500 ppm. SRP Section 6.3 "Emergency Core Cooling System" requires that there not be an unacceptably high concentration of boric acid in the core resulting in precipitation in the long-term cooling phase following a LOCA. The proposed increase in maximum concentration will result in a higher post-LOCA concentration in the recirculated water. However, the post-LOCA concentration resulting from the proposed change is less than the solubility limit. Therefore, precipitation will not occur and the SRP acceptance criteria are satisfied.

The proposed change would lower the minimum SIT boron concentration for Unit 3, Cycle 2 from 1720 to 1420 ppm. SRP Section 4.3 "Nuclear Design" requires reactivity control systems to have a combined capability in conjunction with poison addition from the ECCS to reliably control reactivity changes under postulated accident conditions with a margin for stuck control rods. The proposed change would lower the boric acid concentration during reflood following a large break LOCA. In the large break LOCA analysis, conservatively, no credit is taken for control rod insertion. Analysis has confirmed that with 1420 ppm boric acid from the SIT tanks, the reactor will be sub-critical following a postulated LOCA. Therefore, the proposed change satisfies the SRP acceptance criteria.

The proposed change relaxes boric acid concentration and volume requirements for the BAMU tanks, removes the requirements for heat tracing of the chemical and volume control system (CVCS) and verification that the BAMU tank flow paths can deliver 10 gpm to the RCS, and implements more restrictive requirements on boration flow paths. The CVCS provides RCS inventory control and control of RCS boron concentration for control of reactivity. The functional requirements and acceptance criteria for the CVCS are contained in SRP Section 5.4.7 "Residual Heat Removal System" and Section 9.3.4 "Chemical and Volume Control System." SRP Section 5.4.7 by reference to Branch Technical Position 5-1 "Design Requirements of the Residual Heat Removal System" require the capability to take the plant from normal operating conditions to cold shutdown assuming a loss of offsite power and the limiting single failures. SRP Section 9.3.4 requires that the CVCS: 1) be capable of providing negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences; 2) be capable of providing makeup for small breaks to the RCS pressure boundary and to function as part of the ECCS assuming a single active failure coincides with loss of offsite power; 3) be capable of providing boration of the reactor coolant system through either of two flow paths and from either of two sources. In addition, SRP Section 9.3.4 requires that the amount of boric acid stored in the CVCS exceeds the amount required to borate the RCS to cold shutdown conditions assuming the highest worth CEA is held fully withdrawn from the core. To satisfy these requirements, an adequate volume of sufficiently borated water must be available in the BAMU tanks and RWST, and flow paths must be available to inject water into the RCS. The proposed requirements ensure that an adequate volume of borated water and flow paths are available to enable cooldown of the plant assuming letdown is unavailable in conjunction with a loss of offsite power and the limiting single failure. The proposed concentration requirements ensure that sufficient boron is available to maintain shutdown margin during plant cooldown from normal operating to cold shutdown conditions. The proposed boration flow path requirements ensure that borated water is available from either of two sources and that a borated water source is available to function as part of the ECCS assuming a single active failure coincident with a loss of offsite power. Therefore, the proposed change meets the SRP acceptance criteria. Based on the above, the proposed changes are similar to Example (vi) in that they may, in some way, reduce existing requirements but where the changes satisfy the SRP acceptance criteria.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; and (2) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed changes; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:4954F

ATTACHMENT "A"

Unit 2 Existing Specifications

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from either boric acid makeup tank via either one of the boric acid makeup pumps, the blending tee or the gravity feed connection and any charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

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 above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Flow paths from one or both boric acid makeup tanks via

   1. The associated gravity feed connection(s) and/or

   2. The associated boric acid makeup pump(s)

   via charging pump(s) to the RCS

and/or

b. The flow path from the refueling water storage tank via charging pump(s) to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3.0% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two 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 from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

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.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a SIAS test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 40 gpm to the Reactor Coolant System.

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage tanks with:

  1. A minimum borated water volume of 9970 gallons above the ECCS suction connection,

  2. A minimum boron concentration of 1720 ppm, and

  3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

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 of the tank, and

  3. Verifying the boric acid makeup 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 when the outside air temperature is less than 40°F or greater than 100°F.

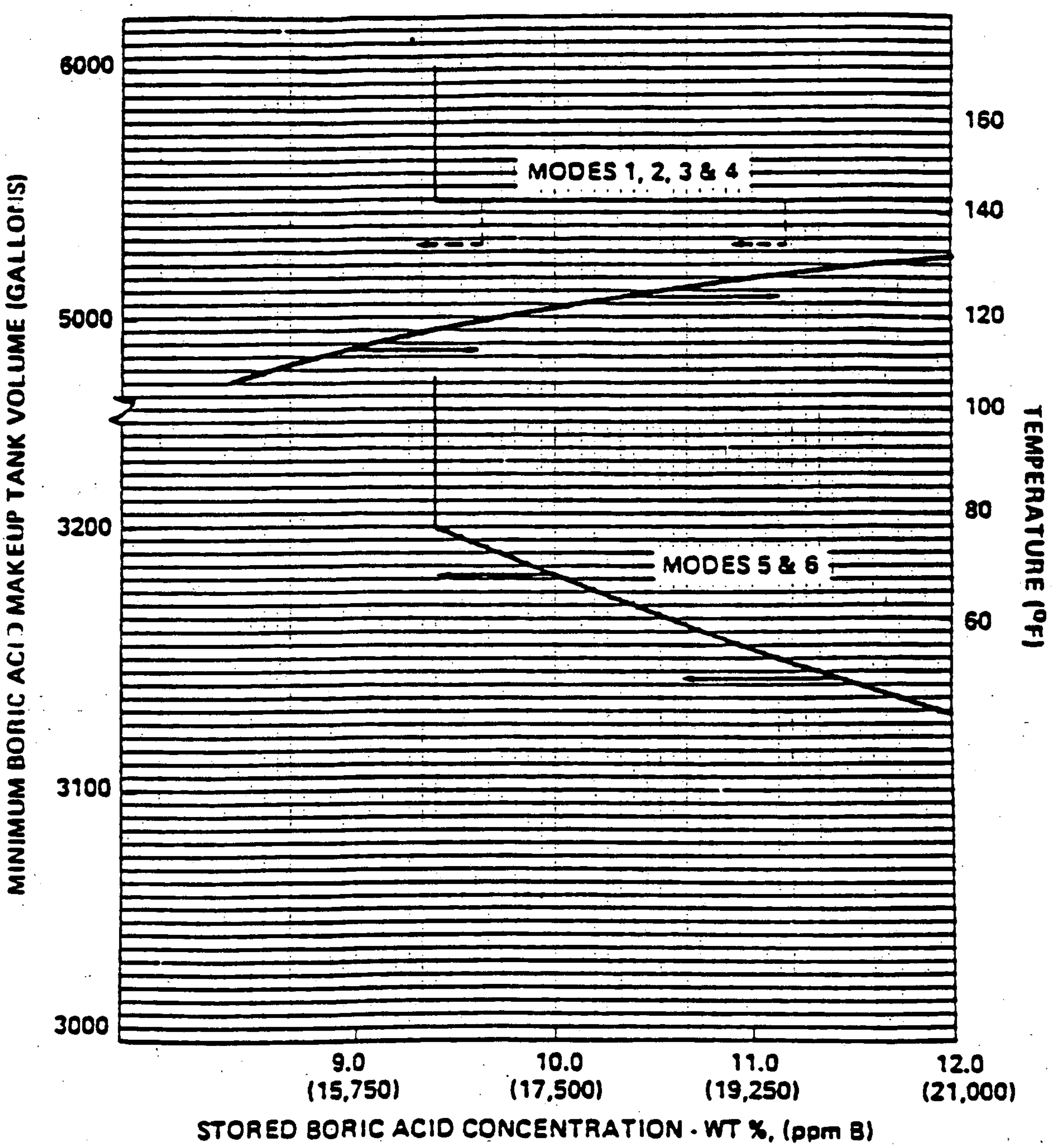


Figure 3.1-1
MINIMUM BORIC ACID STORAGE TANK VOLUME AND TEMPERATURE AS A FUNCTION OF STORED BORIC ACID CONCENTRATION

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. At least one boric acid makeup tank and at least one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,

2. Between 1720 and 2300 ppm of boron, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water 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 sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

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

3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 100°F.

DEC 19 1984

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 and 2300 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

---

*With pressurizer pressure greater than or equal to 715 psia.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum borated water volume of 362,800 gallons above the ECCS suction connection,

b. Between 1720 and 2300 ppm of boron, and

c. A solution temperature between 40°F and 100°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

4.5.4 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 when the outside air temperature is less than 40°F or greater than 100°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°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 an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup 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 SHUTDOWN MARGIN from expected operating conditions of 3.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 81,970 gallons of 2000 ppm borated water from the refueling water tank. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to the more restrictive value of Specification 3.5.4.

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 changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 3% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 9,970 gallons of 2000 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

ATTACHMENT "B"

Unit 2 Proposed Specifications

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from either boric acid makeup tank via either one of the boric acid makeup pumps, the blending tee or the gravity feed connection and any charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. Intentionally deleted.

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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flowpaths to the RCS via the charging pump(s) shall be OPERABLE:

a. At least one of the following combinations:

1) One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, its associated gravity feed valve, and boric acid makeup pump,

2) Two boric acid makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-1, their associated gravity feed valves, and boric acid makeup pumps,

3) Two boric acid makeup tanks, each with contents in accordance with Figure 3.1-1, at least one gravity feed valve, and at least one boric acid makeup pump, and

b. The flow path from the refueling water storage tank.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With fewer than the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore the required boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3.0% delta k/k at 200°F within the next 6 hours; restore the required flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

a. Intentionally deleted.

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.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a SIAS test signal.

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 The boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the boric acid makeup pump(s) required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump(s) 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 The boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the boric acid makeup pump(s) required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump(s) 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank with a minimum boron concentration of 1720 ppm and a minimum borated water volume of 5150 gallons, or

b. The refueling water storage tanks with:

1. A minimum borated water volume of 5150 gallons above the ECCS suction connection,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

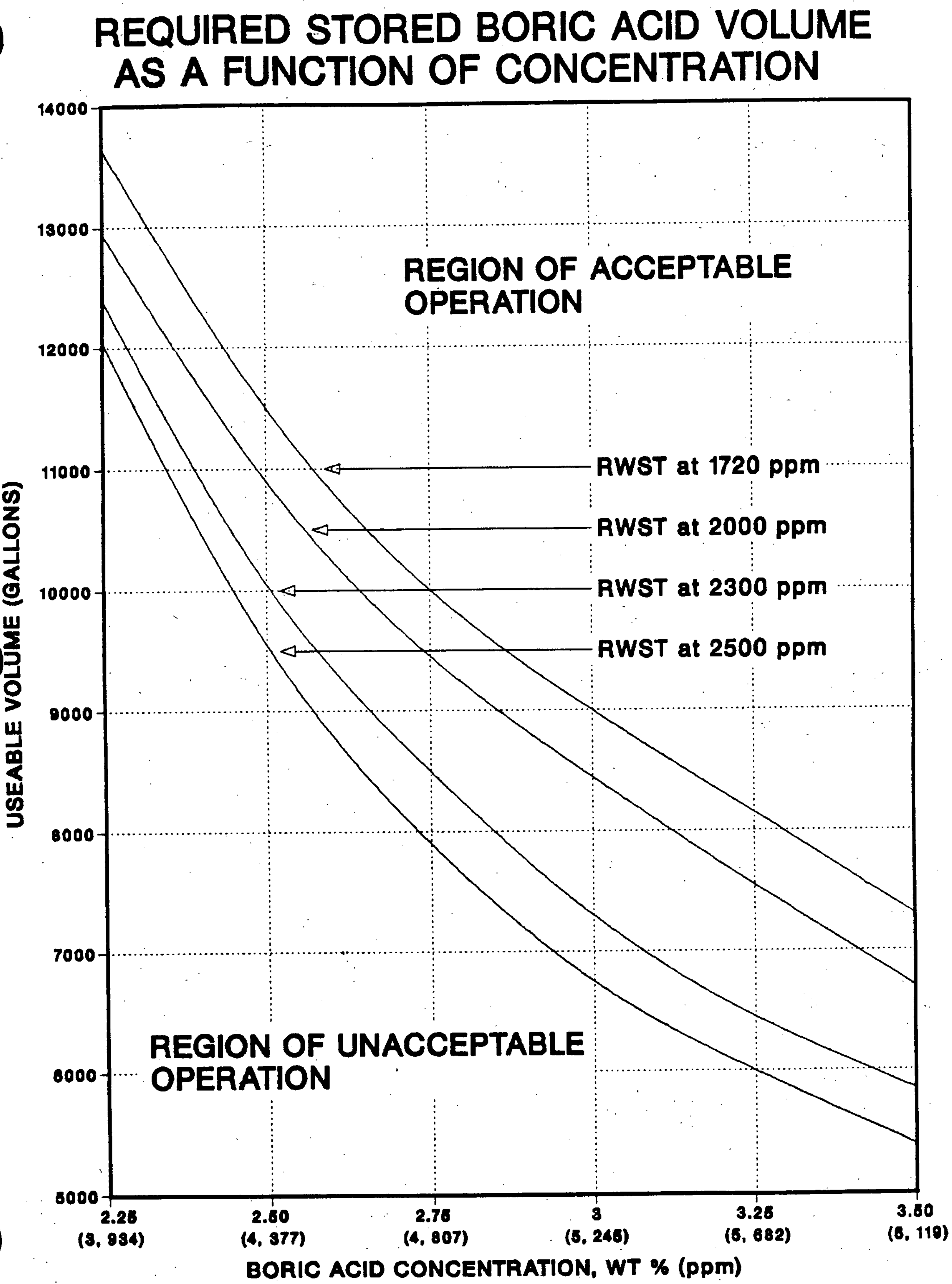
a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and

2. Verifying the contained borated water volume of the tank.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water when the outside air temperature is less than 40°F or greater than 100°F.

Figure 3.1-1


REQUIRED STORED BORIC ACID VOLUME
AS A FUNCTION OF CONCENTRATION
REGION OF ACCEPTABLE OPERATION
RWST at 1720 ppm
RWST at 2000 ppm
RWST at 2300 ppm
RWST at 2500 ppm
REGION OF UNACCEPTABLE OPERATION
USEABLE VOLUME (GALLONS)
14000
13000
12000
11000
10000
9000
8000
7000
6000
5000
2.25 (3, 934)
2.50 (4, 377)
2.75 (4, 807)
3 (5, 245)
3.25 (5, 682)
3.50 (6, 119)
BORIC ACID CONCENTRATION, WT % (ppm)

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 The following borated water sources shall be OPERABLE:

a. At least one of the following combinations:

1) One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, its associated gravity feed valve, and boric acid makeup pump,

2) Two boric acid makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-1, their associated gravity feed valves, and boric acid makeup pumps,

3) Two boric acid makeup tanks, each with contents in accordance with Figure 3.1-1, at least one gravity feed valve, and at least one boric acid makeup pump and,

b. The refueling water storage tank with:

1) A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,

2) Between 1720 and 2500 ppm of boron, and

3) A solution temperature between 40^{o}F and 100^{o}F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the tank(s) 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 3.0% delta k/k at 200^{o}F; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water 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 sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and

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

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40^{o}F or greater than 100^{o}F.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 and 2500ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

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*With pressurizer pressure greater than or equal to 715 psia.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum borated water volume of 362,800 gallons above the ECCS suction connection,

b. Between 1720 and 2500 ppm of boron, and

c. A solution temperature between 40°F and 100°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

4.5.4 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 when the outside air temperature is less than 40°F or greater than 100°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°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 an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup pumps, and 5) 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 SHUTDOWN MARGIN from expected operating conditions of 3.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 plus approximately 13,000 gallons of 1720 ppm borated water from the refueling water tank or approximately 45,000 gallons of 1720 ppm borated water from the refueling water tank alone. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to more restrictive value of Specification 3.5.4.

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 changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 3% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5150 gallons of 1720 ppm borated water from either the refueling water tank or boric acid solution from a boric acid makeup tank.

ATTACHMENT "C"

Unit 3 Existing Specifications

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from either boric acid makeup tank via either one of the boric acid makeup pumps, the blending tee or the gravity feed connection and any charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

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 above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup 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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

a. Flow paths from one or both boric acid makeup tanks via

1. The associated gravity feed connection(s) and/or

2. The associated boric acid makeup pump(s)

via charging pump(s) to the RCS

and/or

b. The flow path from the refueling water storage tank via charging pump(s) to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3.0% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two 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 from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

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.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a SIAS test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 40 gpm to the Reactor Coolant System.

BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage tanks with:

1. A minimum borated water volume of 9970 gallons above the ECCS suction connection,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

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 of the tank, and

3. Verifying the boric acid makeup 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 when the outside air temperature is less than 40°F or greater than 100°F.

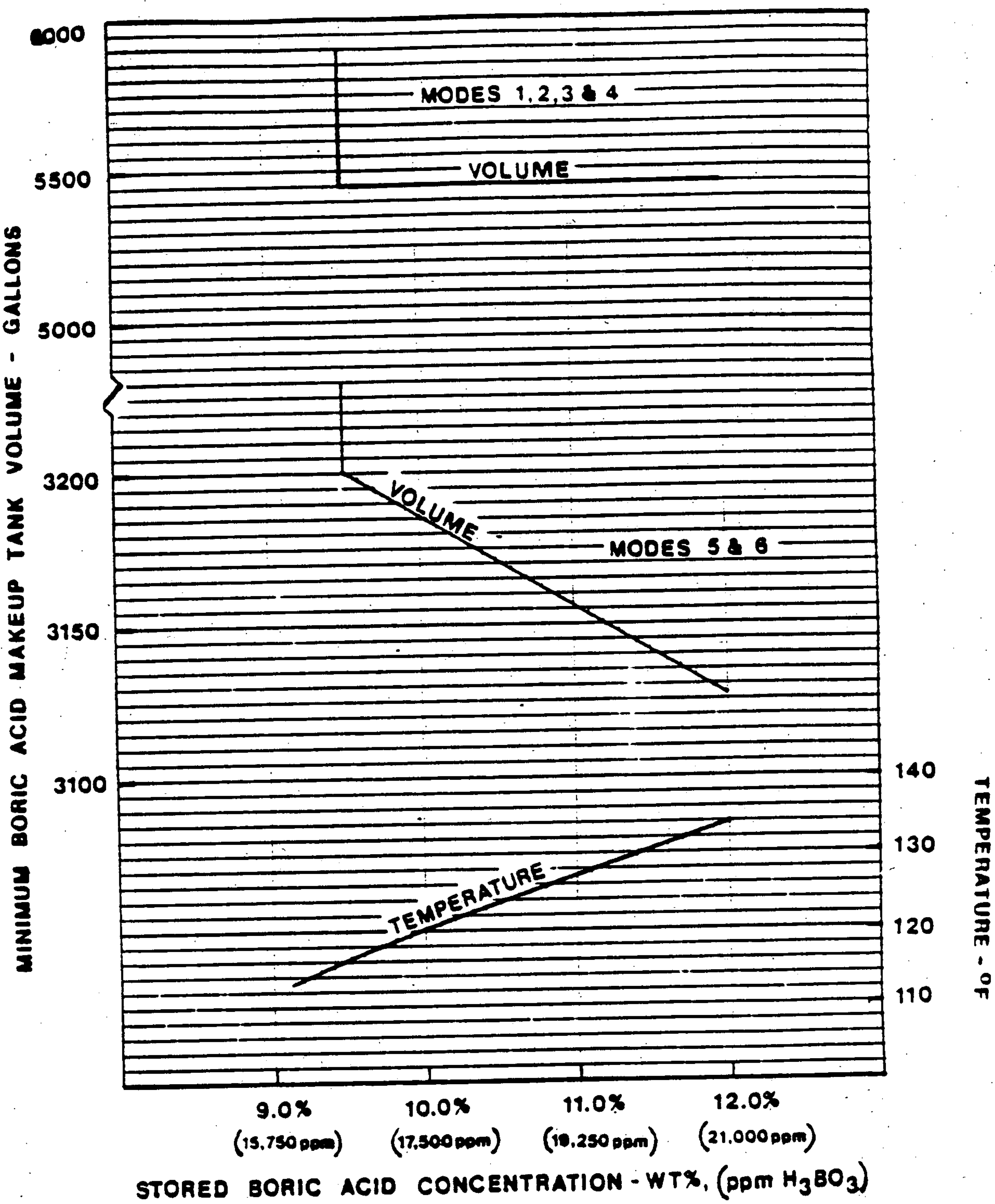

Figure 3.1-1

MINIMUM BORIC ACID STORAGE TANK VOLUME AND MINIMUM TEMPERATURE BEFORE PRECIPITATION AS A FUNCTION OF STORED BORIC ACID H_3BO_3 CONCENTRATION

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. At least one boric acid makeup tank and at least one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water storage tank with:

   1. A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,

   2. Between 1720 and 2300 ppm of boron, and

   3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration in the water,

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

   3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 100°F.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 and 2300 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

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*With pressurizer pressure greater than or equal to 715 psia.

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum borated water volume of 362,800 gallons above the ECCS suction connection,

b. Between 1720 and 2300 ppm of boron, and

c. A solution temperature between 40°F and 100°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

4.5.4 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 when the outside air temperature is less than 40°F or greater than 100°F.

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3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°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 an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup 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 SHUTDOWN MARGIN from expected operating conditions of 3.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 91970 gallons of 1720 ppm borated water from the refueling water tank. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to the more restrictive value of Specification 3.5.4.

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 changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 3.0% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 9970 gallons of 1720 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

ATTACHMENT "D"

Unit 3 Proposed Specifications

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

a. A flow path from either boric acid makeup tank via either one of the boric acid makeup pumps, the blending tee or the gravity feed connection and any charging pump to the Reactor Coolant System if the boric acid makeup tank in Specification 3.1.2.7.a is OPERABLE, or

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. Intentionally deleted.

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.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flowpaths to the RCS via the charging pump(s) shall be OPERABLE:

a. At least one of the following combinations:

1) One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, its associated gravity feed valve, and boric acid makeup pump,

2) Two boric acid makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-1, their associated gravity feed valves, and boric acid makeup pumps,

3) Two boric acid makeup tanks, each with contents in accordance with Figure 3.1-1, at least one gravity feed valve, and at least one boric acid makeup pump, and

b. The flow path from the refueling water storage tank.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With fewer than the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore the required boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 3.0% delta k/k at 200°F within the next 6 hours; restore the required flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

a. Intentionally deleted.

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.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a SIAS test signal.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank with a minimum boron concentration of 1720 ppm and a minimum borated water volume of 5150 gallons, or

b. The refueling water storage tanks with:

1. A minimum borated water volume of 5150 gallons above the ECCS suction connection,
2. A minimum boron concentration of 1720 ppm, and
3. A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

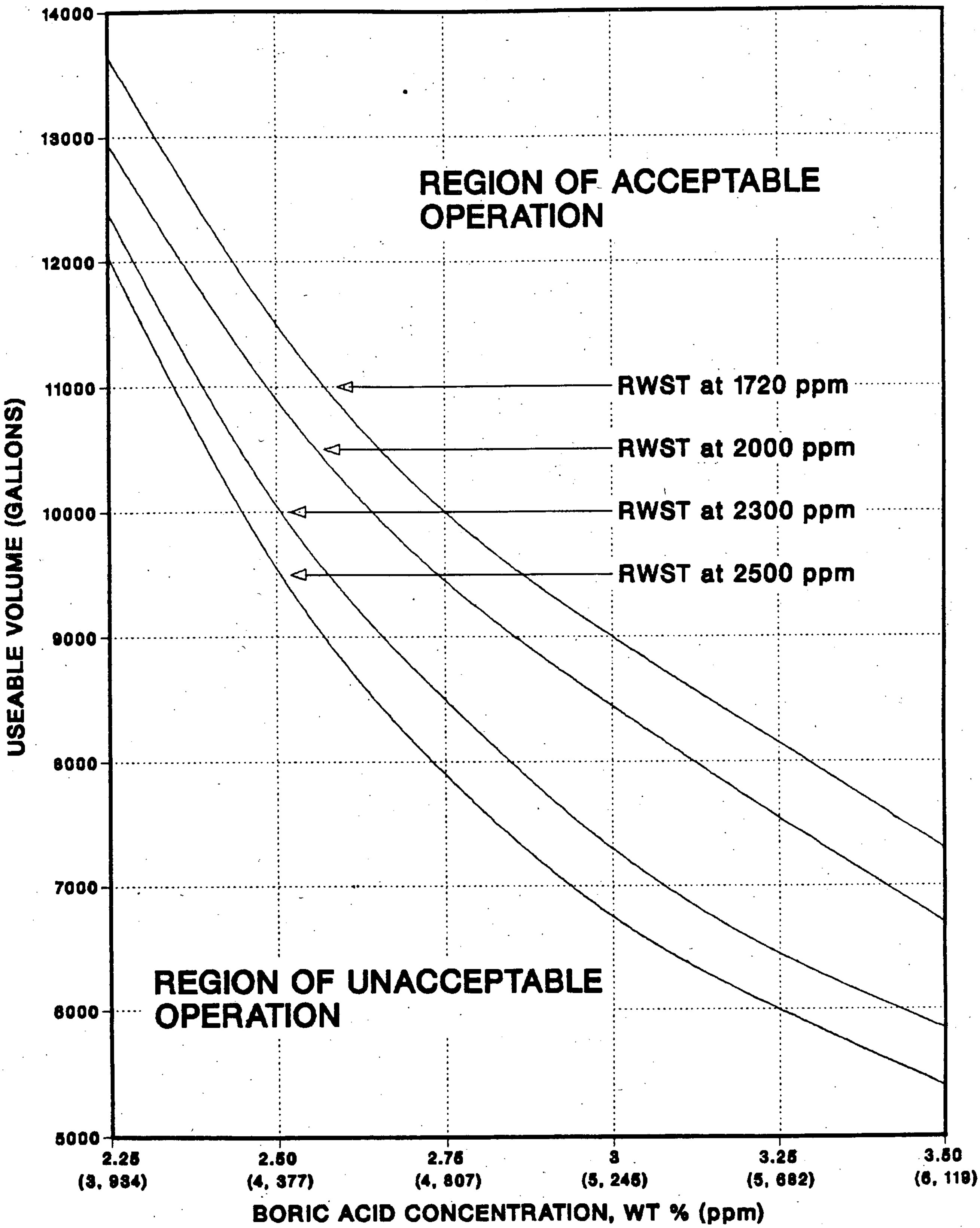
4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water, and
2. Verifying the contained borated water volume of the tank.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water when the outside air temperature is less than 40°F or greater than 100°F.

REQUIRED STORED BORIC ACID VOLUME AS A FUNCTION OF CONCENTRATION



BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 The following borated water sources shall be OPERABLE:

a. At least one of the following combinations:

1) One boric acid makeup tank, with the tank contents in accordance with Figure 3.1-1, its associated gravity feed valve, and boric acid makeup pump,

2) Two boric acid makeup tanks, with the combined contents of the tanks in accordance with Figure 3.1-1, their associated gravity feed valves, and boric acid makeup pumps,

3) Two boric acid makeup tanks, each with contents in accordance with Figure 3.1-1, at least one gravity feed valve, and at least one boric acid makeup pump and,

b. The refueling water storage tank with:

1) A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,

2) Between 1720 and 2500 ppm of boron, and

3) A solution temperature between 40°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the tank(s) 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 3.0% delta k/k at 200°F; restore the above required boric acid makeup tank(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water 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 sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and

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

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 100°F.

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 (1420 for Cycle 2) and 2500 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure greater than or equal to 715 psia.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

a. A minimum borated water volume of 362,800 gallons above the ECCS suction connection,

b. Between 1720 and 2500 ppm of boron, and

c. A solution temperature between 40°F and 100°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

4.5.4 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 when the outside air temperature is less than 40°F or greater than 100°F.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 520°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 an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid makeup pumps, and 5) 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 SHUTDOWN MARGIN from expected operating conditions of 3.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 plus approximately 13,000 gallons of 1720 ppm borated water from the refueling water tank or approximately 45,000 gallons of 1720 ppm borated water from the refueling water tank alone. However, for the purpose of consistency the minimum required volume of 362,800 gallons above ECCS suction connection in Specification 3.1.2.8 is identical to more restrictive value of Specification 3.5.4.

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 changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 3% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5150 gallons of 1720 ppm borated water from either the refueling water tank or boric acid solution from a boric acid makeup tank.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-201 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.2.4, "DNBR Margin", and 3/4.3.1, "Reactor Protective Instrumentation", and their associated bases.

Existing Specifications

Unit 2: See Attachment "A"
Unit 3: See Attachment "C"

Proposed Specifications

Unit 2: See Attachment "B"
Unit 3: See Attachment "D"

Description

The proposed change revises Technical Specifications 3/4.2.4, "DNBR Margin", and 3/4.3.1, "Reactor Protective Instrumentation". Technical Specification 3/4.2.4 requires that the departure from nucleate boiling ratio (DNBR) be maintained by operating within the region of acceptable operation as indicated by either the Core Operating Limit Supervisory System (COLSS) or the Core Protection Calculator (CPC). Technical Specification 3/4.3.1 requires that the Reactor Protective Instrumentation System (RPIS) be operable and defines the number and type of RPIS channels required, response times, periodic testing required to assure operability, and actions to be taken when the required RPIS is out of service.

The proposed change consists of the following four parts:

a) The proposed change revises Figures 3.2-1, "DNBR Operating Limit Based on COLSS", Figure 3.2-2, "DNBR Margin Operating Limit Based on Core Protection Calculators (COLSS out of service, with power greater than or equal to 80% of rated thermal power)", and Figure 3.2-3, "DNBR Margin Operating Limit Based on Core Protection Calculators (COLSS out of service, with power less than 80% of rated thermal power)". Specifically, Figure 3.2-1 shows the minimum calculated Power Operating Limit (POL) based on DNBR for a given reactor power level. Provided that the COLSS POL is greater than that required by Figure 3.2-1 for a given reactor power, the plant is operating in the region of acceptable operation. The existing Figure 3.2-2 defines the required minimum DNBR margin for reactor operation at greater than or equal to 80% power. Figure 3.2-3 defines the required minimum DNBR margin for reactor operation at less than 80% power. The proposed change to Technical Specification 3.2.4 replaces the existing Limiting Condition for Operation (LCO) with four parts, 3.2.4.a through 3.2.4.d. 3.2.4.a and 3.2.4.b replace the existing Figure 3.2-1 with words to the same effect.

3.2.4.a states that when COLSS is in service and either one or both Control Element Assembly Calculators (CEACs) are operable, the COLSS calculated core power must be maintained less than or equal to the COLSS calculated POL based on DNBR. 3.2.4.b states that when COLSS is in service and neither CEAC is operable, the COLSS calculated core power must be maintained less than or equal to the COLSS calculated POL based on DNBR decreased by a penalty factor of 13.0% of rated power. 3.2.4.c states that when COLSS is out of service and either one or both CEACs are operable, CPC calculated DNBR on any operable channel must be kept within the limits of Figure 3.2-1. The new Figure 3.2-1 is a power independent figure replacing the existing Figures 3.2-2 and 3.2-3. Section 3.2.4.d states that when COLSS is out of service and neither CEAC is operable, CPC calculated DNBR on any operable channel must be kept within the limits of Figure 3.2-2. The new Figure 3.2-2 is a power independent figure similar to the new Figure 3.2-1 but accommodating the increased margin required when both CEAC's are inoperable.

b) The proposed change revises Surveillance Requirement 4.2.4.4, which provides the rod bow penalty factors on DNBR as a function of fuel burnup, which are included in the COLSS and the CPC DNBR calculations. Specifically, this Surveillance Requirement requires that the rod bow penalty factor on DNBR as a function of fuel exposure should be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days. The justification for removal of this surveillance requirement is that the rod bow penalty factor has been determined to be less than 2.0 percent at a fuel exposure of 30,000 MWD/MTU. This is unchanged from cycle 2. Because of the physical radial power peak burndown effect, no fuel assembly with burnup exceeding 30,000 MWD/MTU would produce sufficient power to be subject to a limiting DNBR condition. Therefore, 30,000 MWD/MTU is a cutoff point for rod bow penalty calculation. The rod bow penalty factor will be verified for each cycle by design analysis. Since the rod bow penalty at 30,000 MWD/MTU has been incorporated in the minimum DNBR limit of the CPC's, Surveillance Requirement 4.2.4.4 is not required.

c) The proposed change revises Technical Specification 3.3.1, Table 3.3-1, ACTION 6 which provides conditions under which operation may continue with COLSS out of service, for various operability conditions of the CEAC's. ACTION 6.b addresses operation with COLSS out of service and either one or both CEAC's are operable. ACTION 6.c addresses operation when COLSS is out of service and neither CEAC is operable. The proposed Action change to ACTION 6 combines ACTIONs 6.b and 6.c. Except for whether or not the CEAC's are operable, these two ACTIONs are essentially the same. The reference to Figure 3.2-1 in the existing ACTION 6.b is replaced by reference to Specification 3.2.4.b. The reference to the penalty factor on the BERR1 constant in the existing ACTION 6.c is replaced by reference to Specification 3.2.4.d, where the same penalty definition is applied by a new mechanism.

d) The proposed change revises Surveillance Requirement 4.2.4.2 which specifies monitoring of DNBR whenever THERMAL POWER is above 20% of RATED

THERMAL POWER. Specifically, Surveillance Requirement 4.2.4.2 requires that DNBR be determined by continuously monitoring core power distribution using COLSS, or, with COLSS out of service, by verifying every two hours that the DNBR on all OPERABLE CPC system DNBR channels is within the limits of Figures 3.2-2 or 3.2-3, and that the penalties in Table 3.3-2b are implemented. The proposed change to Surveillance Requirement 4.2.4.2 removes the requirement to verify that the appropriate penalty factors have been implemented on all CPC system channels, with a requirement to verify operation to be within the limits of new Figures 3.2-1 or 3.2-2 on any CPC system channel. The proposed change recognizes that it is only necessary to monitor one channel for control purposes during steady state operation, as is done for other parameters. Since sufficient margin is already implemented in the CPC trip setpoint, it is not necessary that this be the most limiting channel. The four CPC channels continue to provide the protection required during transient operation.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any one of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   The simplification of requirements in Technical Specifications 3/4.2.4 and 3/4.3.1 results in a small change in plant operation. This simplification is expected to reduce, rather than increase, the probability of an accident. The deletion of the DNBR rod bow penalty factors from Technical Specification 3/4.2.4 will not result in an increase in the probability or consequences of an accident as the application of the appropriate penalty factors remains adequately validated by off line analysis.

   Standard Review Plan (SRP), Section 4.4, "Fuel System Design", specifies acceptance fuel design limits which must not be exceeded. In addition, SRP Section 7.2 requires that the Reactor Protective Instrumentation System (RPIS) automatically initiates reactor trip to assure that specified acceptable fuel design limits are not exceeded. The results of the reload analysis for events related to these changes are clearly within the established criteria of the SRP and there is no significant increase in the consequences of accidents previously analyzed.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The only changes to plant operation resulting from the proposed changes are: 1) the operators will have fewer figures and requirements to apply, and 2) the operators will no longer need to check that the appropriate DNBR penalty factor has been included in the CPC's. Neither of these changes create a situation which could result in a new or different kind of accident from one previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

Response: No

For burnups less than 30,000 MWD/MTU the appropriate penalty is in place. For burnups greater than 30,000 MWD/MTU no penalty is needed. Therefore, the margin of safety based on the established SRP criteria remains as it was originally assumed.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to a purely administrative change to technical specifications, for example a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature. Example (iii) relates to a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and the NRC has previously found such methods acceptable. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan.

In this case, the proposed change described in (a) above not only revises Figures 3.2-1, Figure 3.2-2, and Figure 3.2-3 but also replaces the existing Limiting Conditions for Operation (LCO) with four parts, i.e., Sections 3.2.4.a through 3.2.4.d. This modification is similar to Example (i) and (iii) as discussed below.

It is similar to Example (i) in that it relates to a purely administrative change to technical specifications by imposing four applicable administrative control methods and two new figures in lieu of six existing figures to maintain an adequate DNBR margin under different states of plant operations.

3.2.4.a and 3.2.4.b replace the existing Figures 3.2-1 with words to the same effect when COLSS is in service. Additionally, both new figures supplant the existing Figures 3.2-2 and 3.2-3 in compliance with 3.2.4.c and 3.2.4.d when COLSS is out of service. Thus, DNBR will be maintained by 3.2.4.a (or 3.2.4.c) when either one or both CEAC are operable and by 3.2.4.b (or 3.2.4.d) when neither CEAC is operable. Since the proposed change pertains to a revision of graphic representations in LCO with a set of plain administrative control statements easy for understanding and two consolidated figures for simplification, it is a change within the scope contemplated by Example (i).

It is similar to Example (iii) in that it relates to a change resulting from a nuclear reactor core loading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not specifically changed, and the NRC has previously found such methods acceptable. This change requires an updating of the DNBR operating limits with the aid of two new figures as a result of the reload analysis for Cycle 3 which invokes a nuclear reactor core loading. There are no fuel assemblies in Cycle 3 which are significantly different from those found previously acceptable to the NRC for the previous core. The DNBR operating limits are derived in compliance with the acceptance criteria used for the previous core on the basis of the analytical methods which were approved by the NRC and found acceptable. Hence the proposed change is similar to Example (iii) in that the DNBR operating limits are updated because of a new core loading in Cycle 3.

The proposed change described in (b) above revises Surveillance Requirements 4.2.4.4 concerning the rod bow penalty factors on DNBR as a function of fuel burnup. This modification is similar to Example (vi) in that the removal of such penalty factors may reduce in some way a margin of safety, but where the results are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan. Sections 4.4 and 7.2 of the SRP delineates acceptance criteria to maintain fuel integrity.

Specifically, the Core Protection Calculator (CPC) and the Reactor Protective Instrumentation System (RPIS) must assure with high probability that acceptable fuel design limits are not exceeded during normal operation or anticipated operational occurrences (AOOs). The removal of Surveillance Requirement 4.2.4.4 does not imply a reduction in the margin to safety because of the deletion of the DNBR rod bow penalty factors from Technical Specification 3/4.2.4. In fact, effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses and combined with other uncertainty factors at the 95/95 confidence/probability level to define a design DNBR limit. They are adequately validated in the CPC software and verified for Cycle 3. In addition, the RPIS will automatically initiate a reactor trip to prevent Specified Acceptable Fuel Design Limits (SAFDLs) from being exceeded during transients and AOOs. Hence, the proposed change will not result in an increase in the probability or consequences of an accident as the application of the appropriate penalty factors remains adequately validated and the protection of RPIS is available. In short, the relaxation of the existing surveillance requirement will not compromise safety margin and the results of the reload analysis related to this proposed change are further verified to ensure that the established criteria of the SRP are met in view of Example (vi).

The proposed change described in (c) above revises the ACTION statements in Table 3.3-1 of Technical Specification 3.3.1. It is similar to Example (i) in that it is a purely administrative change to delete requirements which have been placed in another Technical Specification. The proposed change to ACTION 6 combines ACTIONs 6.b and 6.c for consistency with respect to the proposed changes described in (a) and (b) above. The existing ACTION 6.b.1 is revised to incorporate Specifications 3.2.4.b and 3.2.4.d whereas the existing ACTION 6.c is deleted because the penalty factor on the BERR1 constant referenced therein is applied by a new mechanism per the proposed change (b). Hence the proposed change is similar to Example (i) in that it relates a change to achieve consistency throughout the technical specifications.

The proposed change described in (d) above revises Surveillance Requirement 4.2.4.2 by removing the verification requirement that the appropriate penalty factors have been implemented on all CPC system channels to ensure conformance of the operating limits on any CPC system channel per Figures 3.2-1 or 3.2-2. It is similar to Example (vi) in that the removal of such a surveillance requirement may reduce in some way a margin of safety, but where the results are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan. Specifically, the proposed change recognizes that it is only necessary to monitor one channel for control purposes during steady state operation so long as the plant is operated within the limits of Figures 3.2-1 and 3.2-2 as described in the proposed change (a). Since sufficient margin is already implemented in the CPC trip setpoint for transient protection in a similar way discussed in the proposed change (b), it is not necessary to monitor with all CPC system channels. This proposed change thus relieves the operator of an unnecessarily frequent surveillance without any reduction in a margin to safety. The CPC trips will continue to provide the same protection that they always have during transient operation. Hence the proposed change is similar to Example (vi) in that it is a relaxation of a surveillance requirement without compromising any safety operation.

Safety and Significant Hazards Determination

Based on the Safety Evaluation, it is concluded that: (1) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:4756F

ATTACHMENT A

Existing Technical Specifications, Unit 2

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable and the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

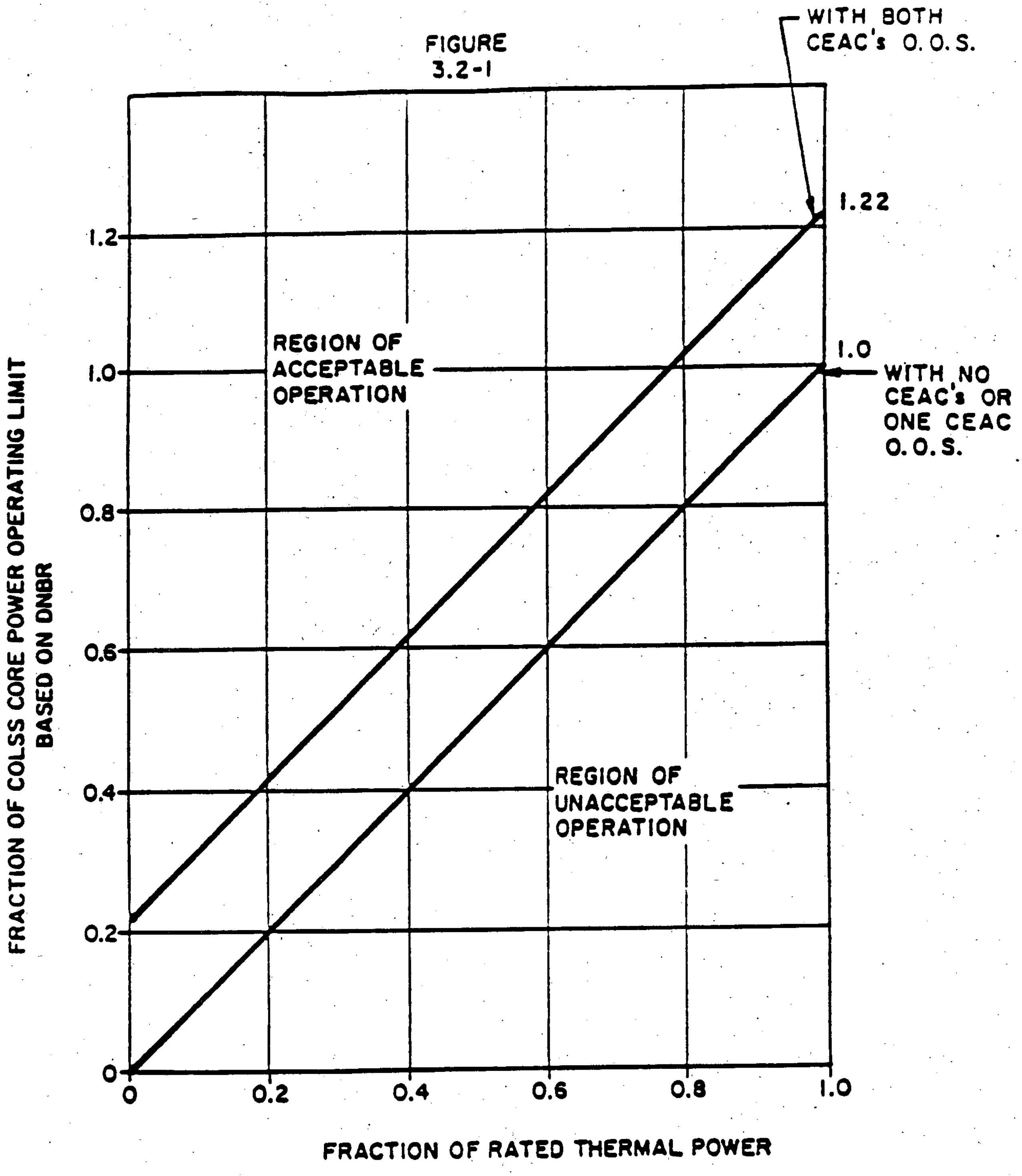


FIGURE 3.2-1 CYCLE 1 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

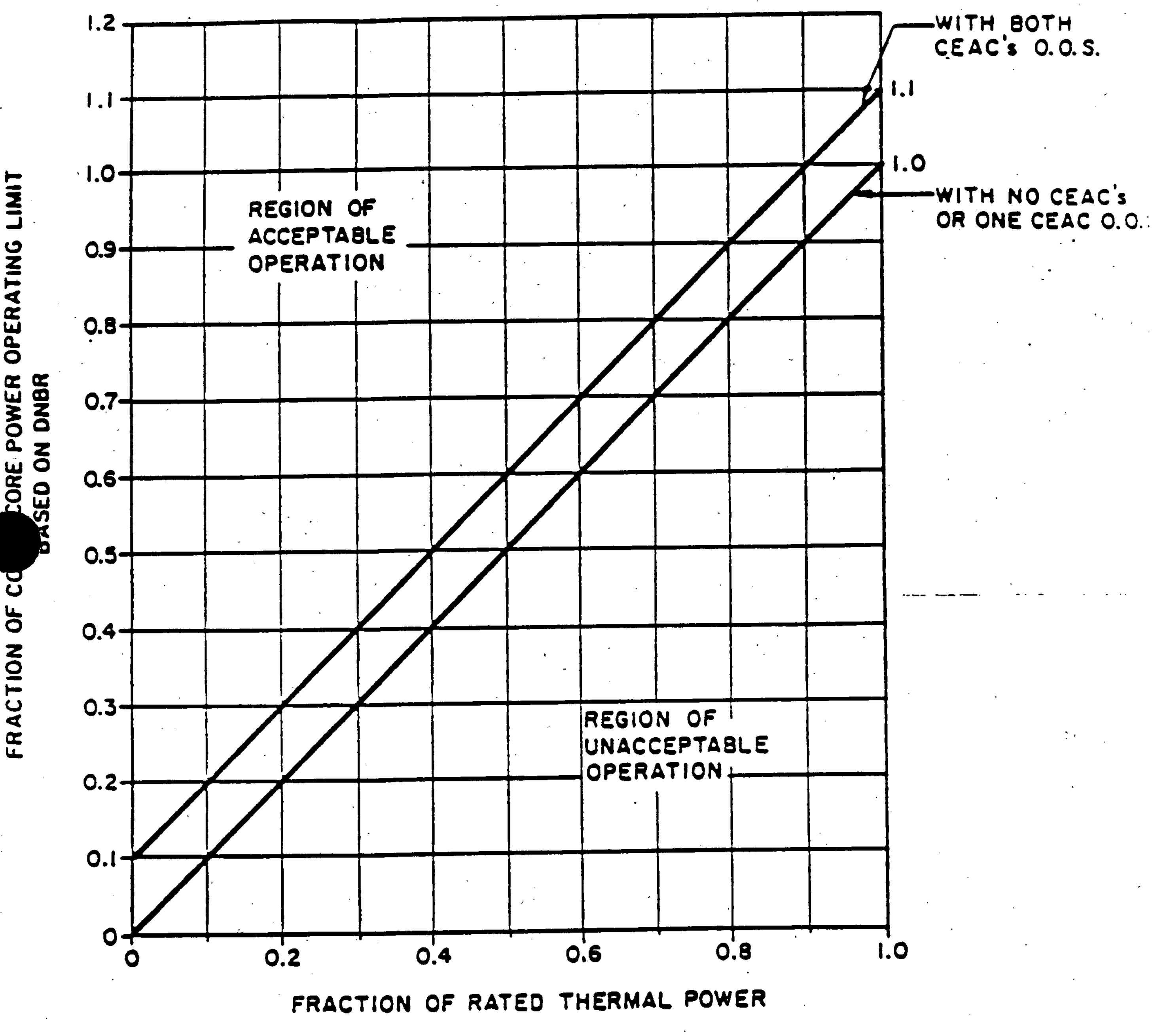


FIGURE 3.2-1 CYCLE 2 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

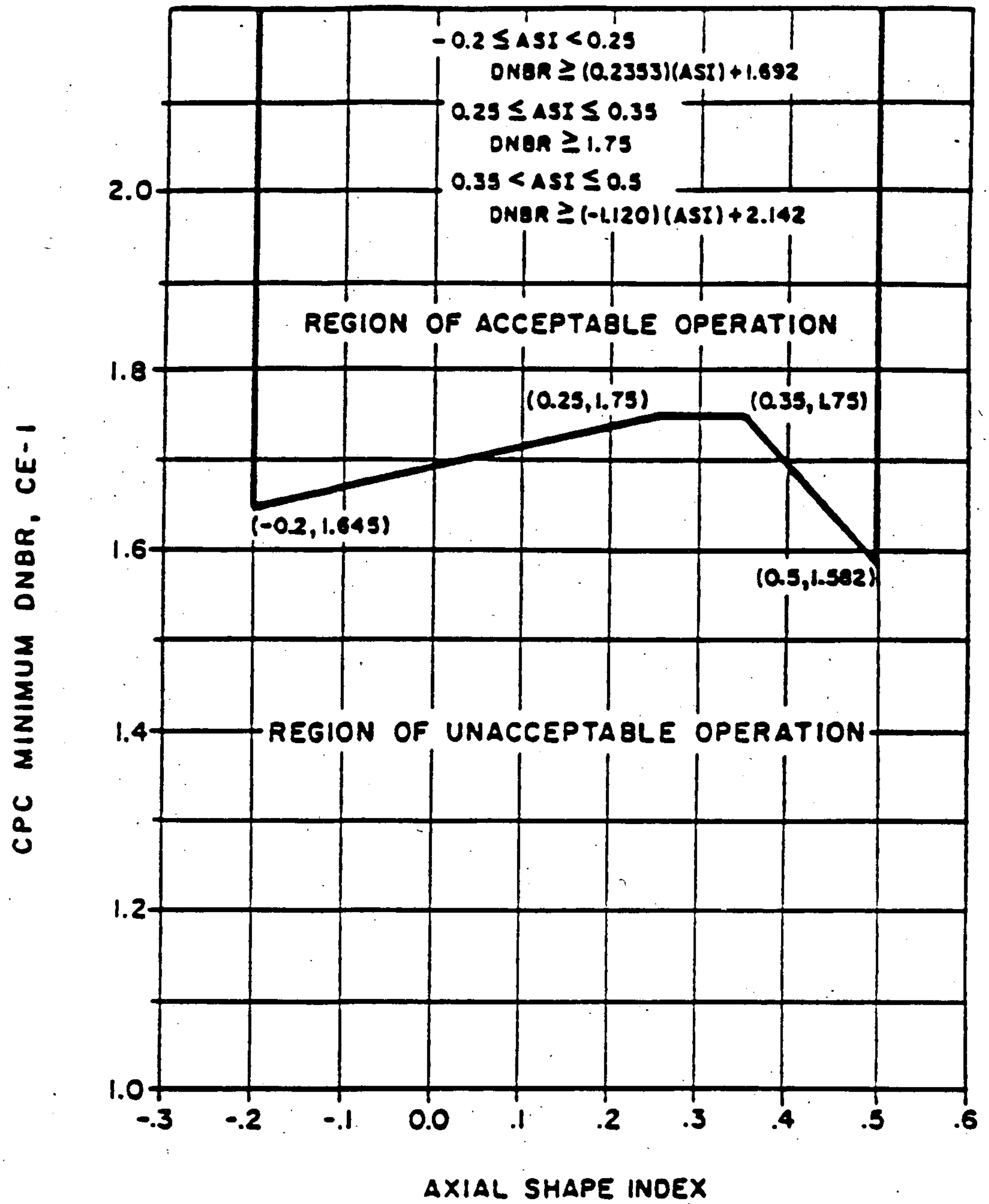


FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 80%)

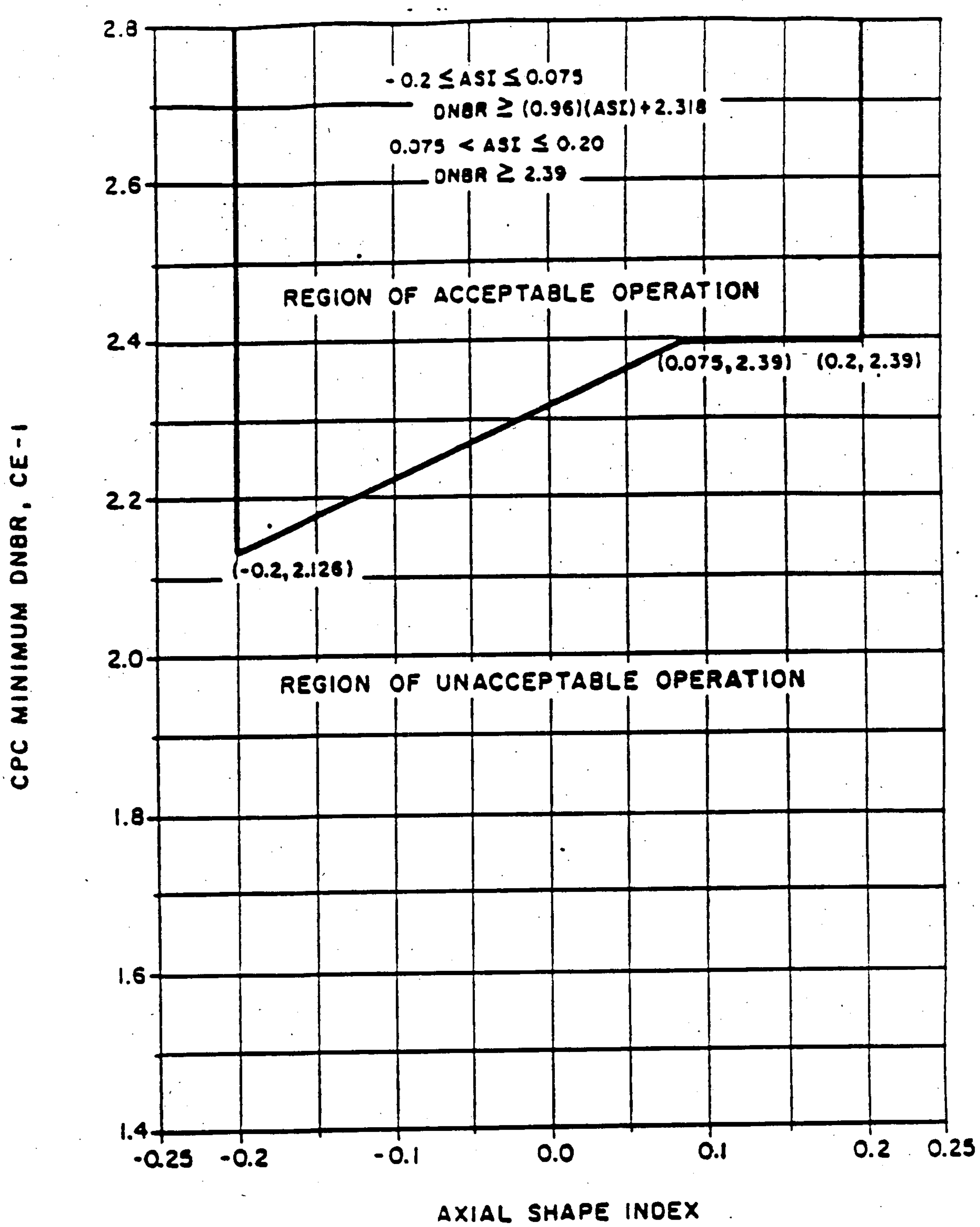


FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 70%)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 -
a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that Action Items 6.b.1, .2 and .3 are met with COLSS in-service, or Action Items 6.c.1, .2 and .3 are met with COLSS out-of-service*.

b. With both CEAC's inoperable and COLSS in-service, operation may continue provided that:*

1. Within 1 hour the DNBR margin operating limit required by Specification 3.2.4 (Figure 3.2-1) is satisfied for both CEAC's out-of-service.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE 3.3-1 (Continued)

TABLE NOTATION

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

c. With both CEAC's inoperable and COLSS out-of-service, operation may continue provided that:*

1. Within 1 hour multiply the CPC value of BERR1 corresponding to COLSS in-service by 1.13 (CYCLE 1) or 1.05 (CYCLE 2) and re-enter into the CPC's.

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE NOTATION

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A - 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.

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 or 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Uncertainty terms already taken into account in the CPC's safety monitoring are removed from Figures 3.2-2 and 3.2-3 since the curves are intended to monitor only the LCO during steady state operation.

DNBR Margin (Continued)

The DNBR penalty factors listed in section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

ATTACHMENT B

Proposed Technical Specifications, Unit 2

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both CEACs are operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13.0% RATED THERMAL POWER (when COLSS is in service and neither CEAC is operable): or

c. Operating within the region of acceptable operation of Figure 3.2-1 using any operable CPC channel (when COLSS is out of service and either one or both CEACs are operable); or

d. Operating within the region of acceptable operation of Figure 3.2-2 using any operable CPC channel (when COLSS is out of service and neither CEACs is operable).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With the DNBR margin not being maintained, as indicated by:

(1) COLSS calculated core power exceeding the appropriate COLSS calculated operating limit, or

(2) With COLSS out of service, operation outside the region of acceptable operation of Figure 3.2-1 or 3.2-2,

Within 15 minutes initiate corrective action to restore the DNBR to within its limits, and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on any OPERABLE DNBR channel, is within the limit shown on Figures 3.2-1 or 3.2-2, as applicable.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

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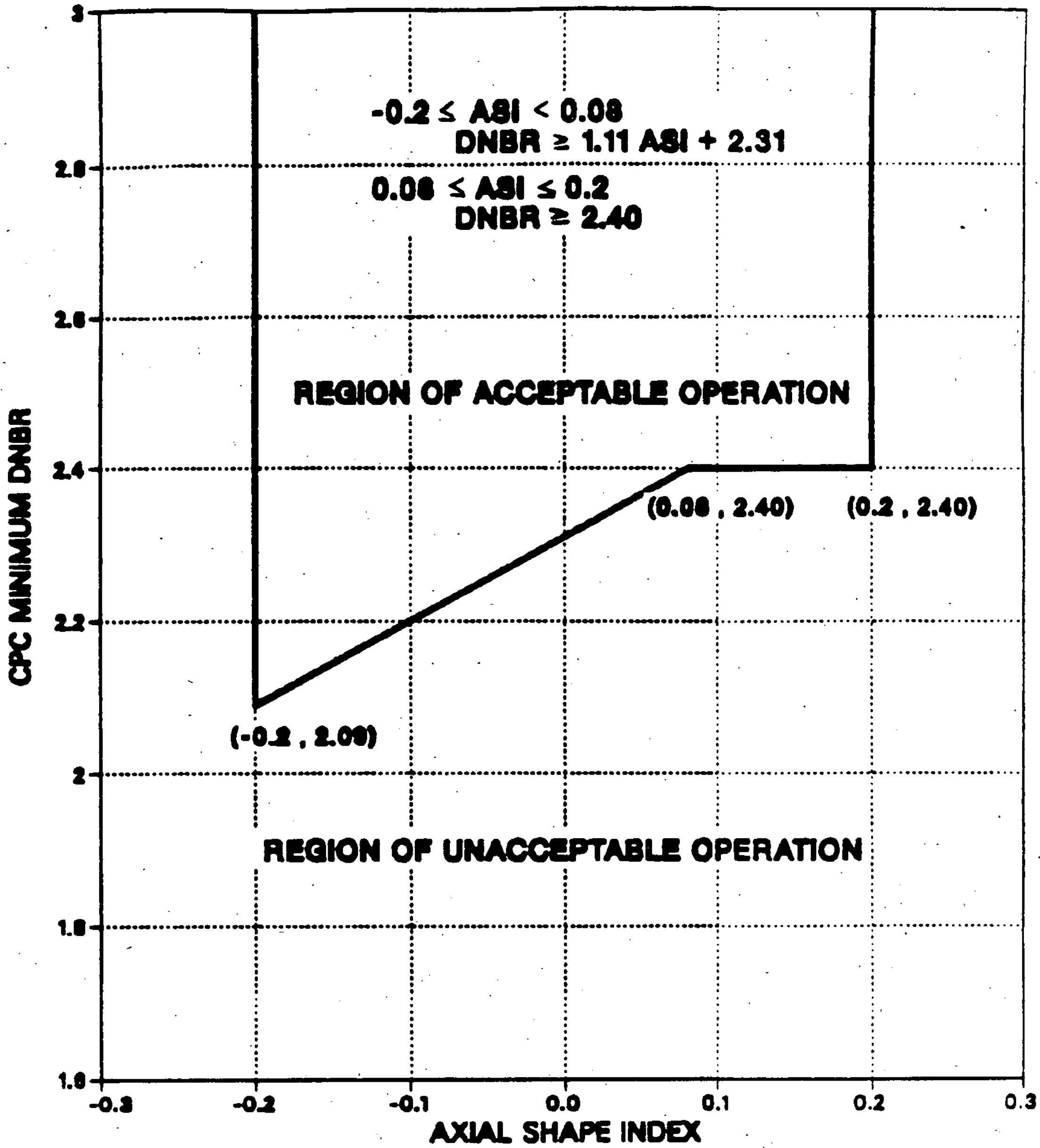

FIGURE 3.2-1 DNBR OPERATING LIMIT BASED
ON CORE PROTECTION CALCULATORS
- COLSS OUT OF SERVICE
- ONE OR BOTH CEACS OPERABLE

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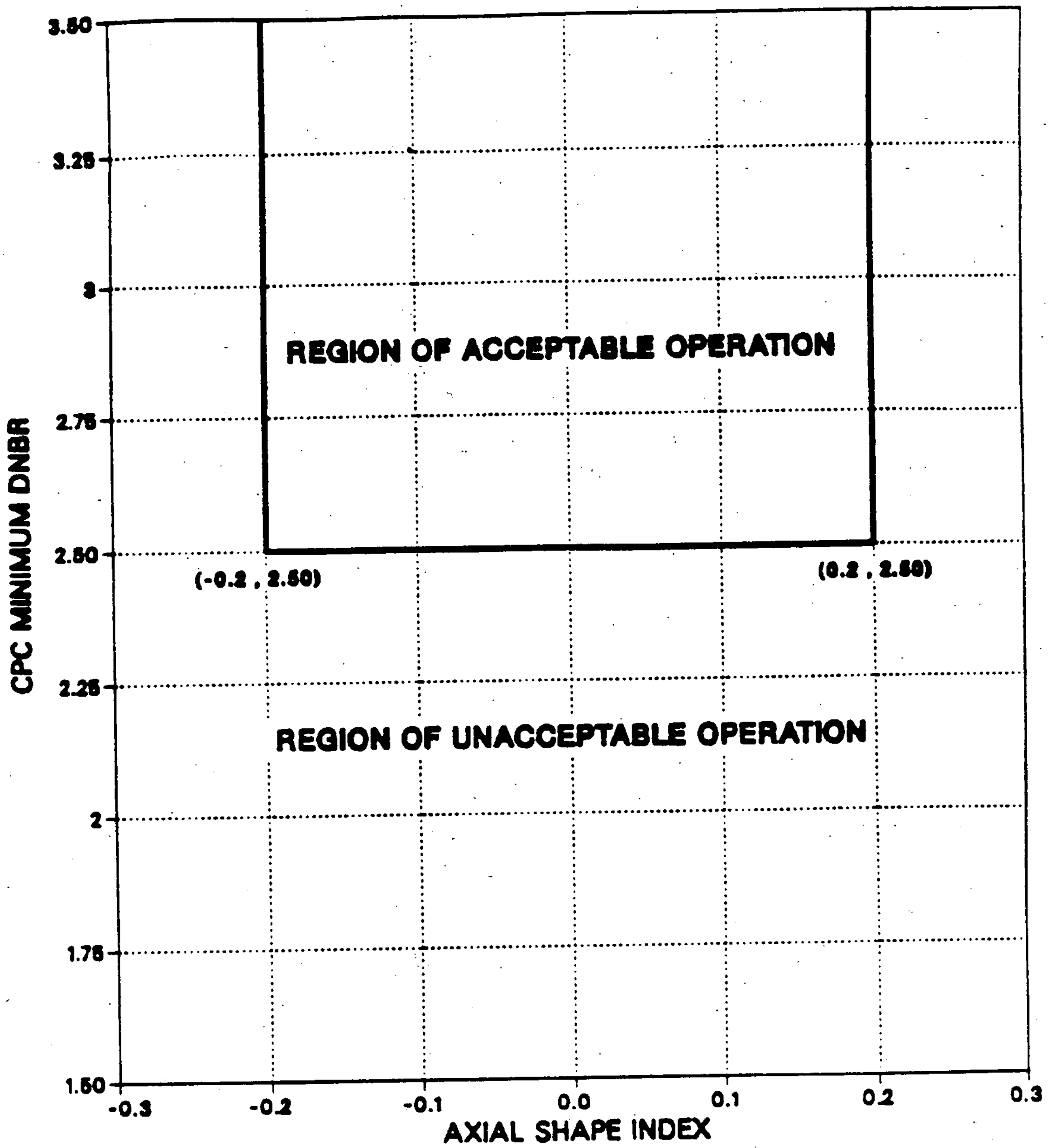

FIGURE 3.2-2 DNBR OPERATING LIMIT BASED ON CORE PROTECTION CALCULATOR
- COLSS OUT OF SERVICE
- BOTH CEACS INOPERABLE

DELETED INTENTIONALLY

DELETED INTENTIONALLY

DELETED INTENTIONALLY

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that ACTION 6.b is met.*

b. With both CEACs inoperable, operation may continue provided that:*

1. Within 1 hour the DNBR margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

---

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE 3.3-1 (Continued)

TABLE NOTATION

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A - 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.

DELETED INTENTIONALLY

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minimum DNBR limit) is conservative with respect to the actual core power limit. The penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. In the event that the COLSS is not being used, the DNBR margin can be maintained by monitoring with any operable CPC channel so that the DNBR remains above the predetermined limit as a function of Axial Shape Index. The above listed uncertainty penalty factors are also included in the CPCs, which assume a minimum of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshhold is due to the excore neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The additional uncertainty terms taken into account in the CPCs for transient protection are removed from Figures 3.2-1 and 3.2-2 since the curves are intended to monitor the LCO only during steady state operation.

DNBR Margin (continued)

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each fuel assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

ATTACHMENT C

Existing Technical Specifications, Unit 3

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable and the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The DNBR penalty factors included in the COLSS and CPC DNBR calculations shall be verified at least once per 31 EFPDs to be greater than or equal to the values listed below. This verification will be made on the basis of the BERR1 addressable constant for the CPC and the EPOL2 addressable constant for the COLSS.

Burnup $\frac{GWD}{MTU}$	DNBR Penalty (%)
0-10	0.5
10-20	1.0
20-30	2.0
30-40	3.5
40-50	5.5

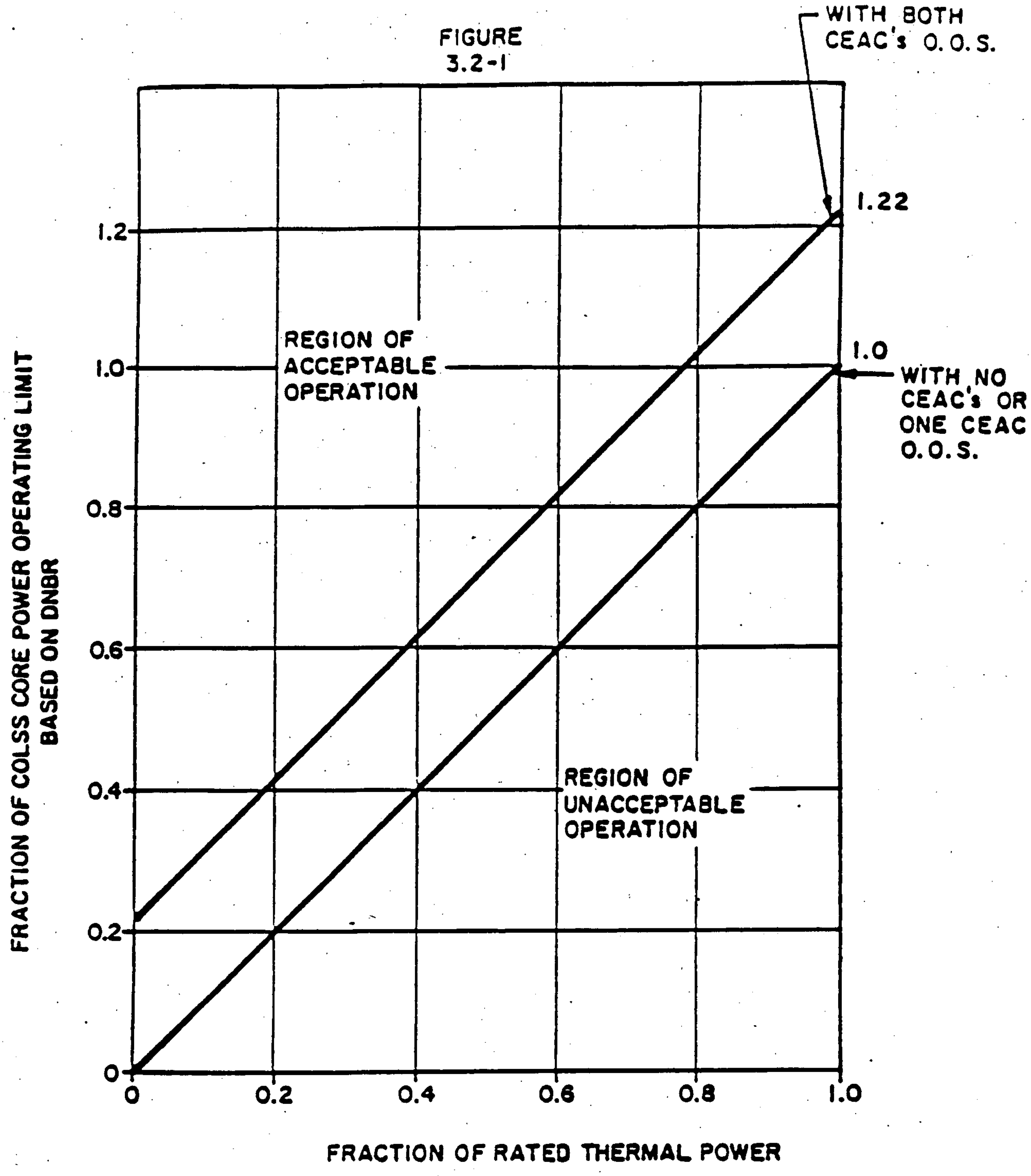

FIGURE 3.2-1 CYCLE 1 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

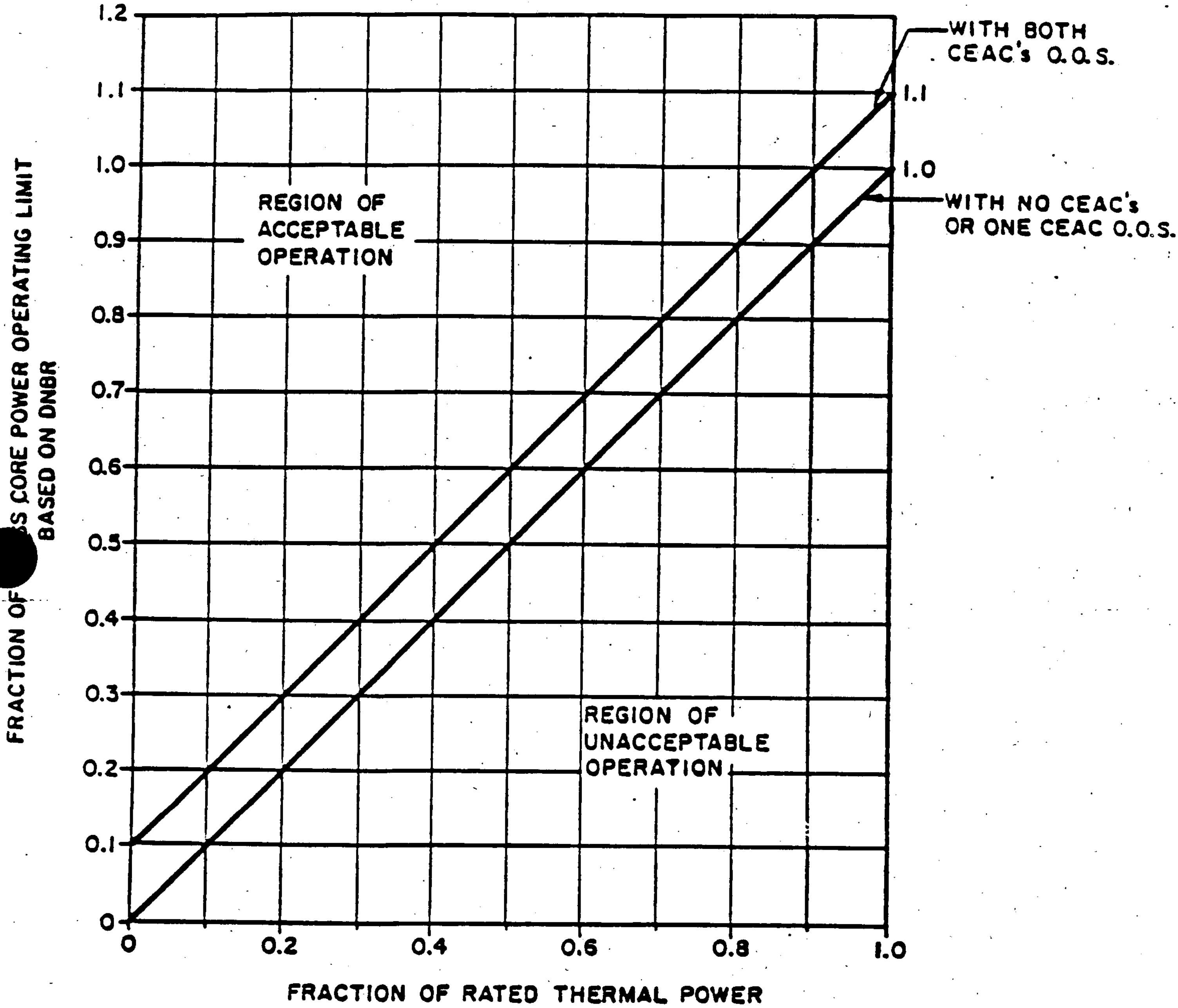

FIGURE 3.2-1 CYCLE 2 DNBR MARGIN OPERATING LIMIT BASED ON COLSS.

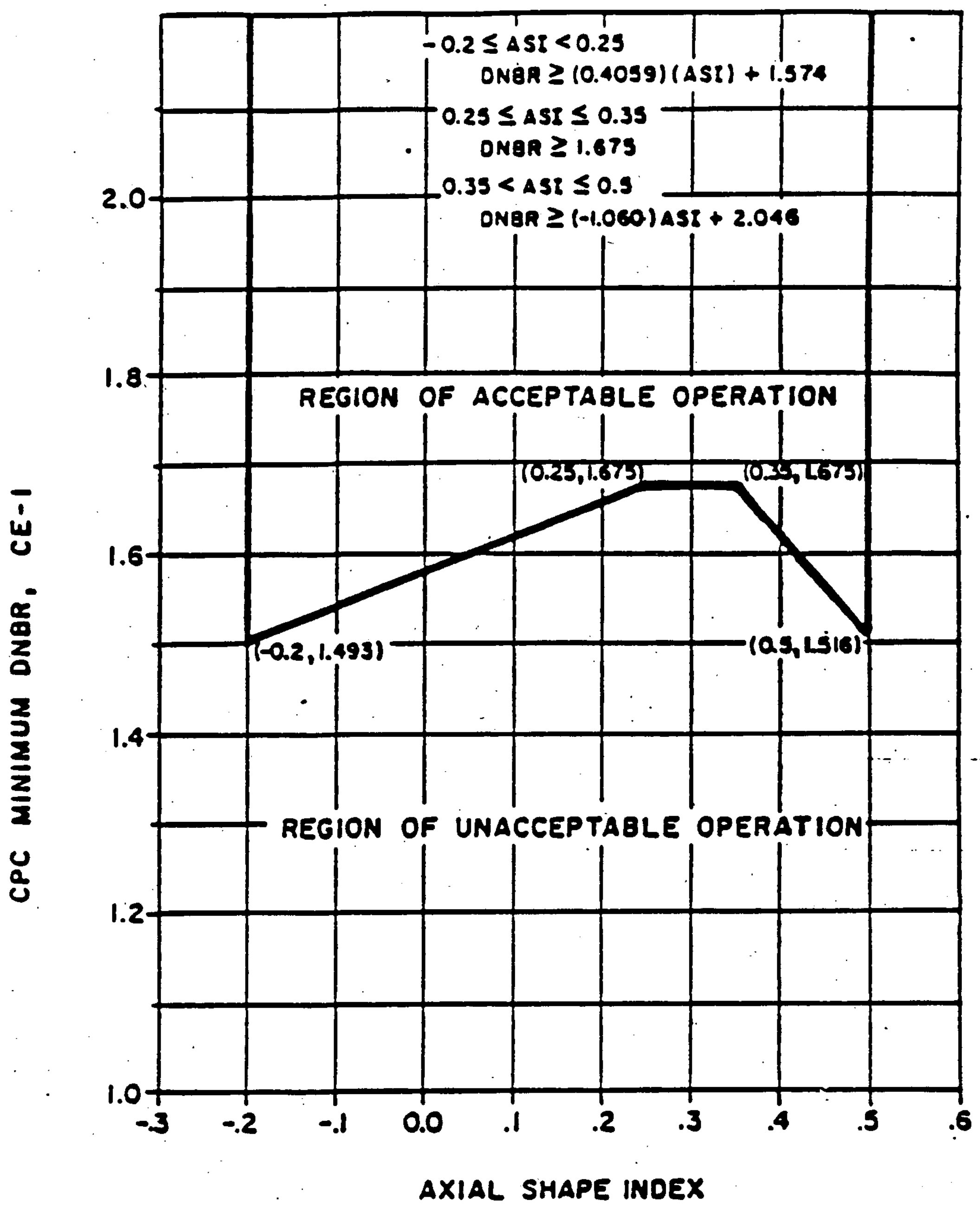

FIGURE 3.2-2 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER ≥ 80%)

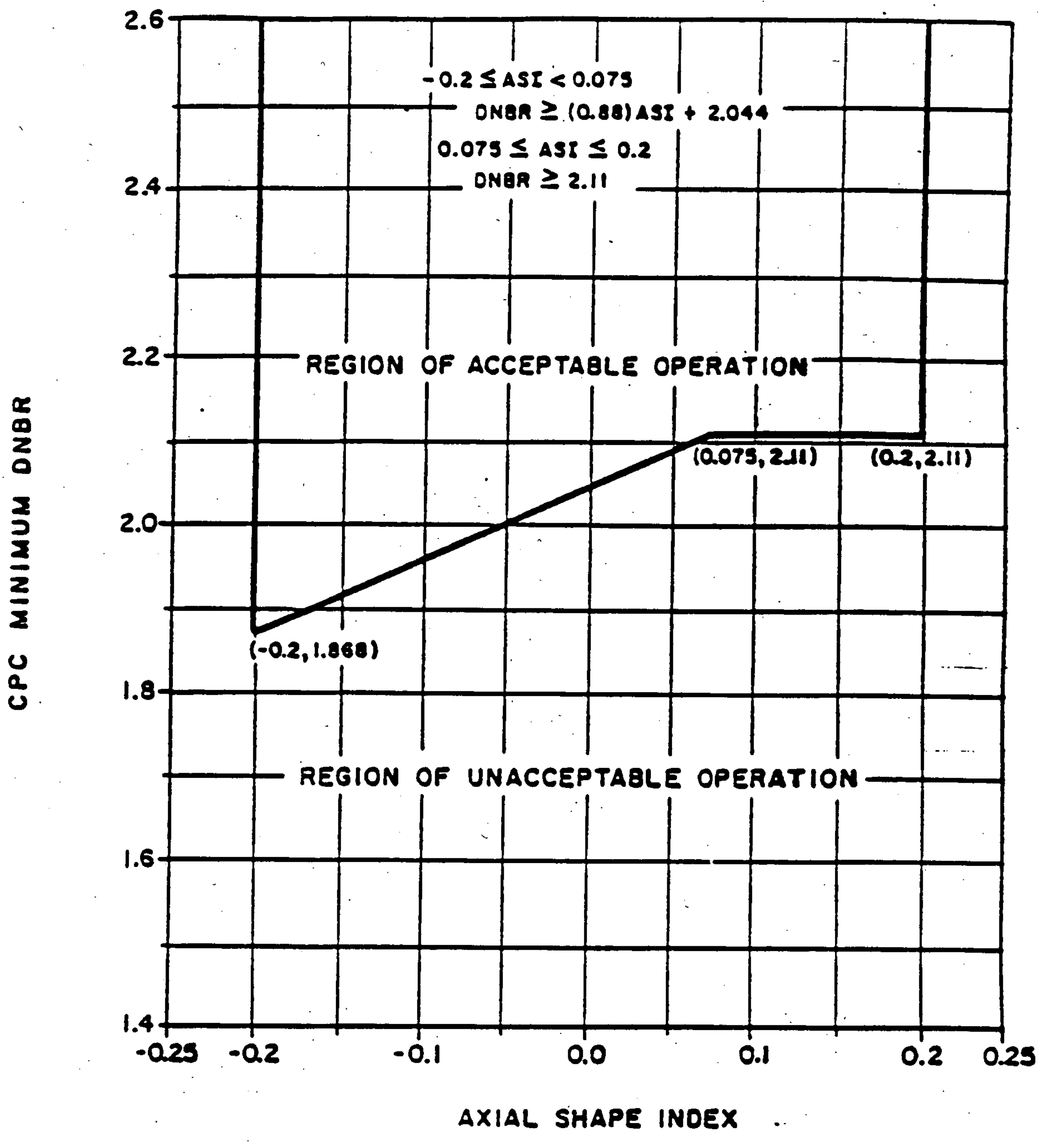

FIGURE 3.2-2 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER ≥ 70%)

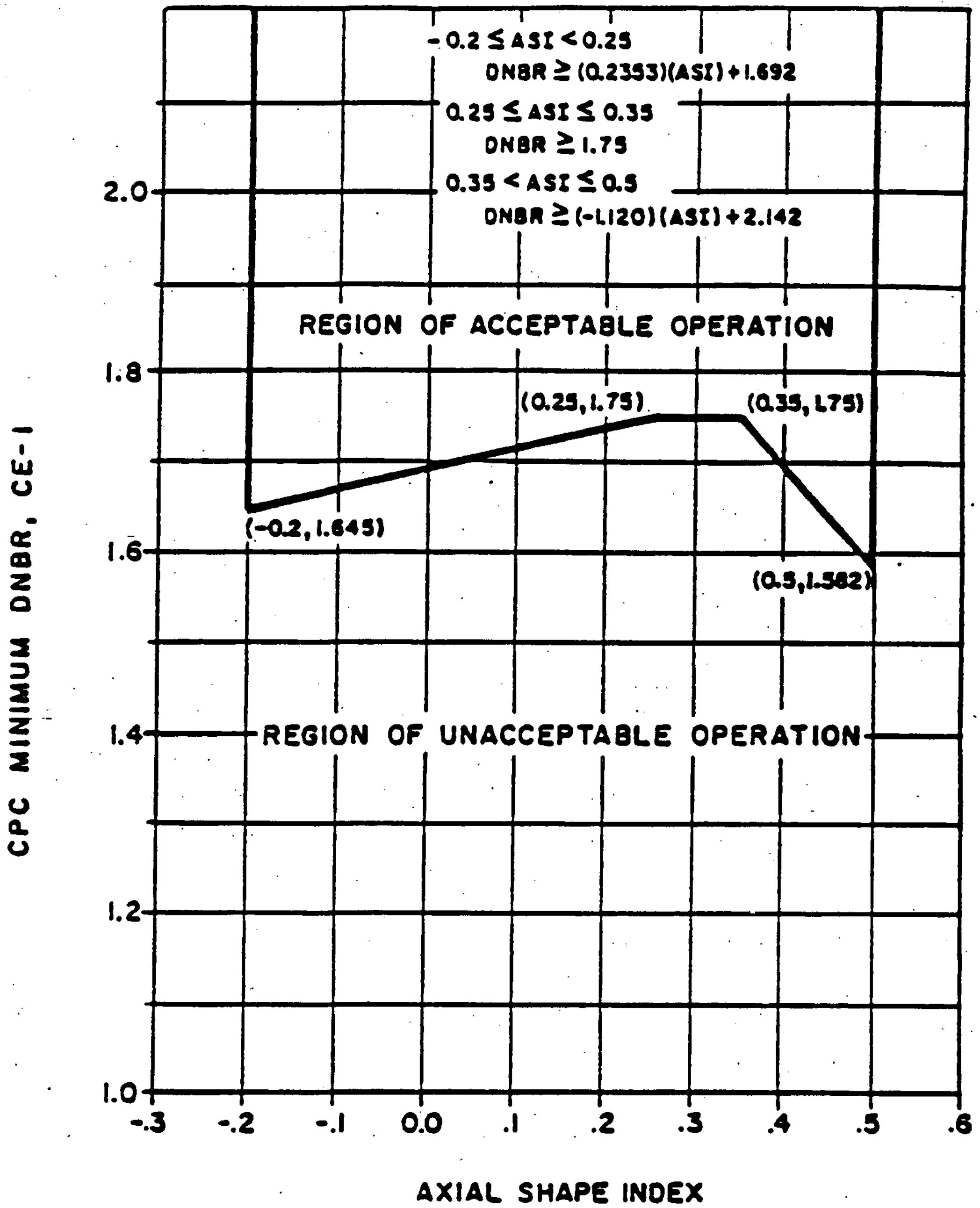

FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 1, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 80%)

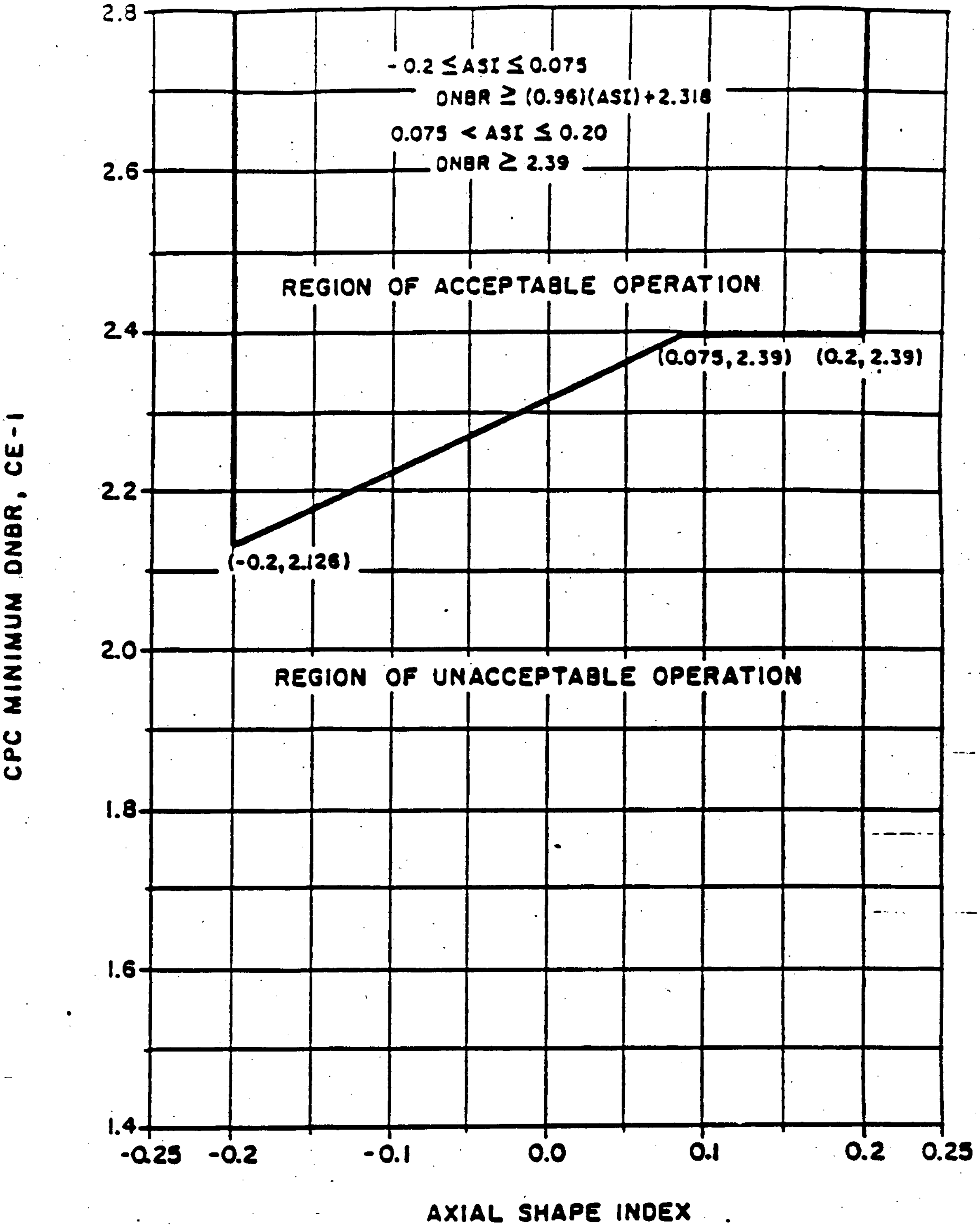

FIGURE 3.2-3 DNBR MARGIN OPERATING LIMIT BASED ON CORE PROTECTION CALCULATORS (CYCLE 2, COLSS OUT OF SERVICE, WITH RATED THERMAL POWER < 70%)

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that Action Items 6.b.1, .2 and .3 are met with COLSS in-service, or Action Items 6.c.1, .2 and .3 are met with COLSS out-of-service*.

b. With both CEAC's inoperable and COLSS in-service, operation may continue provided that:*

1. Within 1 hour the DNBR margin operating limit required by Specification 3.2.4 (Figure 3.2-1) is satisfied for both CEAC's out-of-service.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

c. With both CEAC's inoperable and COLSS out-of-service operation may continue provided that:*

1. Within 1 hour multiply the CPC value of BERR1 corresponding to COLSS in-service by 1.13 (CYCLE 1) or 1.05 (CYCLE 2) and re-enter into the CPC's.

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2

TABLE 3.3-1 (Continued)

TABLE NOTATION

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A - 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.

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-1 are not violated. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minumum DNBR limit) is conservative with respect to the actual core power limit. These penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-2 or 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty penalty factors plus those associated with startup test acceptance criteria are also included in the CPC's which assume a minimum core power of 20% of RATED THERMAL POWER. The 20% Rated Thermal Power threshold is due to the neutron flux detector system being inaccurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. Uncertainty terms already taken into account in the CPC's safety monitoring are removed from Figures 3.2-2 and 3.2-3 since the curves are intended to monitor only the LCO during steady state operation.

BASES

The DNBR penalty factors listed in Section 4.2.4.4 are penalties used to accommodate the effects of rod bow. The amount of rod bow in each assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. The penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

ATTACHMENT D

Proposed Technical Specifications, Unit 3

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by one of the following methods:

a. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR (when COLSS is in service, and either one or both CEACs are operable); or

b. Maintaining COLSS calculated core power less than or equal to COLSS calculated core power operating limit based on DNBR decreased by 13.0% RATED THERMAL POWER (when COLSS is in service and neither CEAC is operable): or

c. Operating within the region of acceptable operation of Figure 3.2-1 using any operable CPC channel (when COLSS is out of service and either one or both CEACs are operable); or

d. Operating within the region of acceptable operation of Figure 3.2-2 using any operable CPC channel (when COLSS is out of service and neither CEACs is operable).

APPLICABILITY: **MODE 1 above 20% of RATED THERMAL POWER.**

ACTION:

With the DNBR margin not being maintained, as indicated by:

(1) COLSS calculated core power exceeding the appropriate COLSS calculated operating limit, or

(2) With COLSS out of service, operation outside the region of acceptable operation of Figure 3.2-1 or 3.2-2,

Within 15 minutes initiate corrective action to restore the DNBR to within its limits, and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on any OPERABLE DNBR channel, is within the limit shown on Figures 3.2-1 or 3.2-2, as applicable.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

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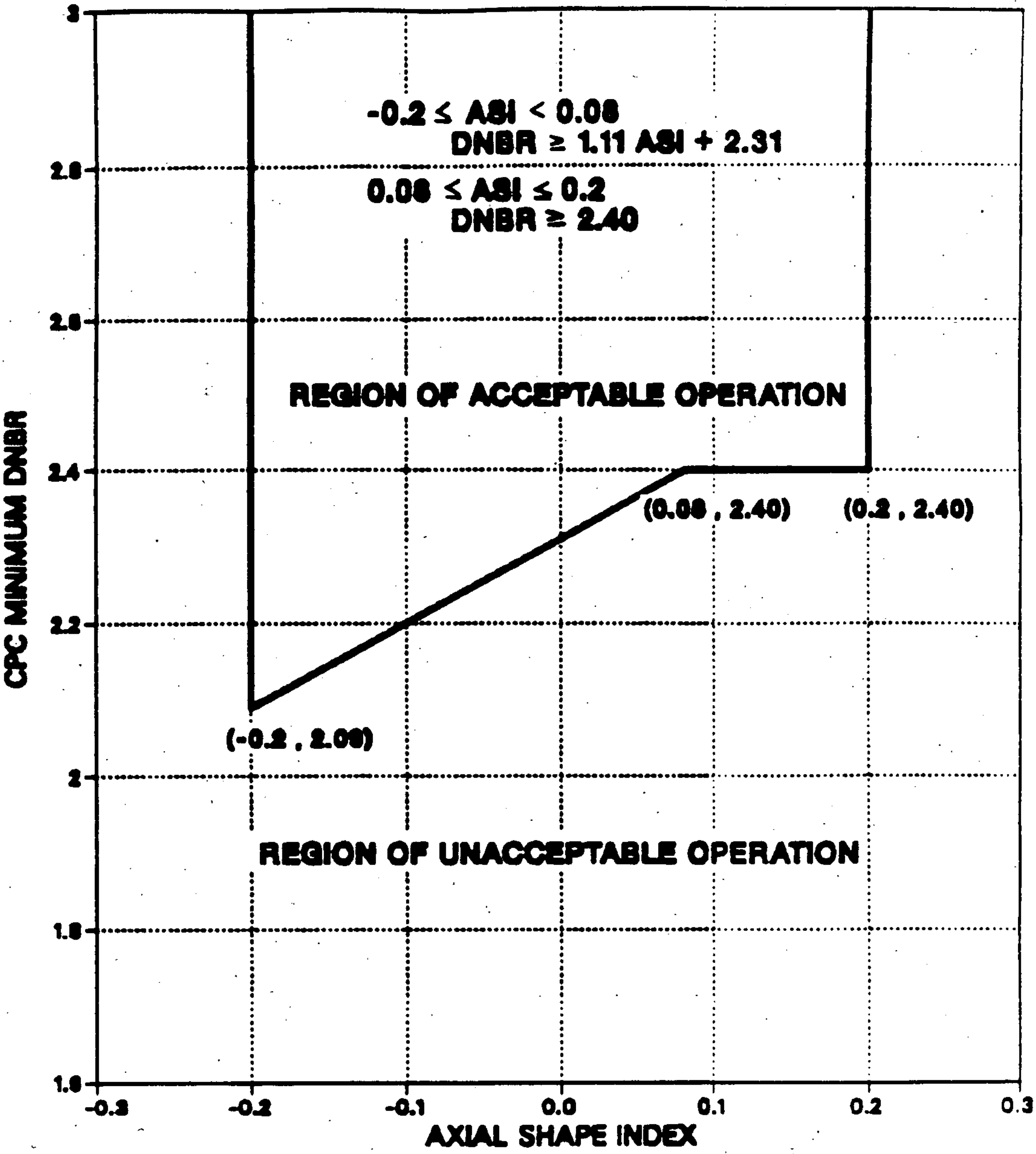

FIGURE 3.2-1 DNBR OPERATING LIMIT BASED
ON CORE PROTECTION CALCULATORS
- COLSS OUT OF SERVICE
- ONE OR BOTH CEACS OPERABLE

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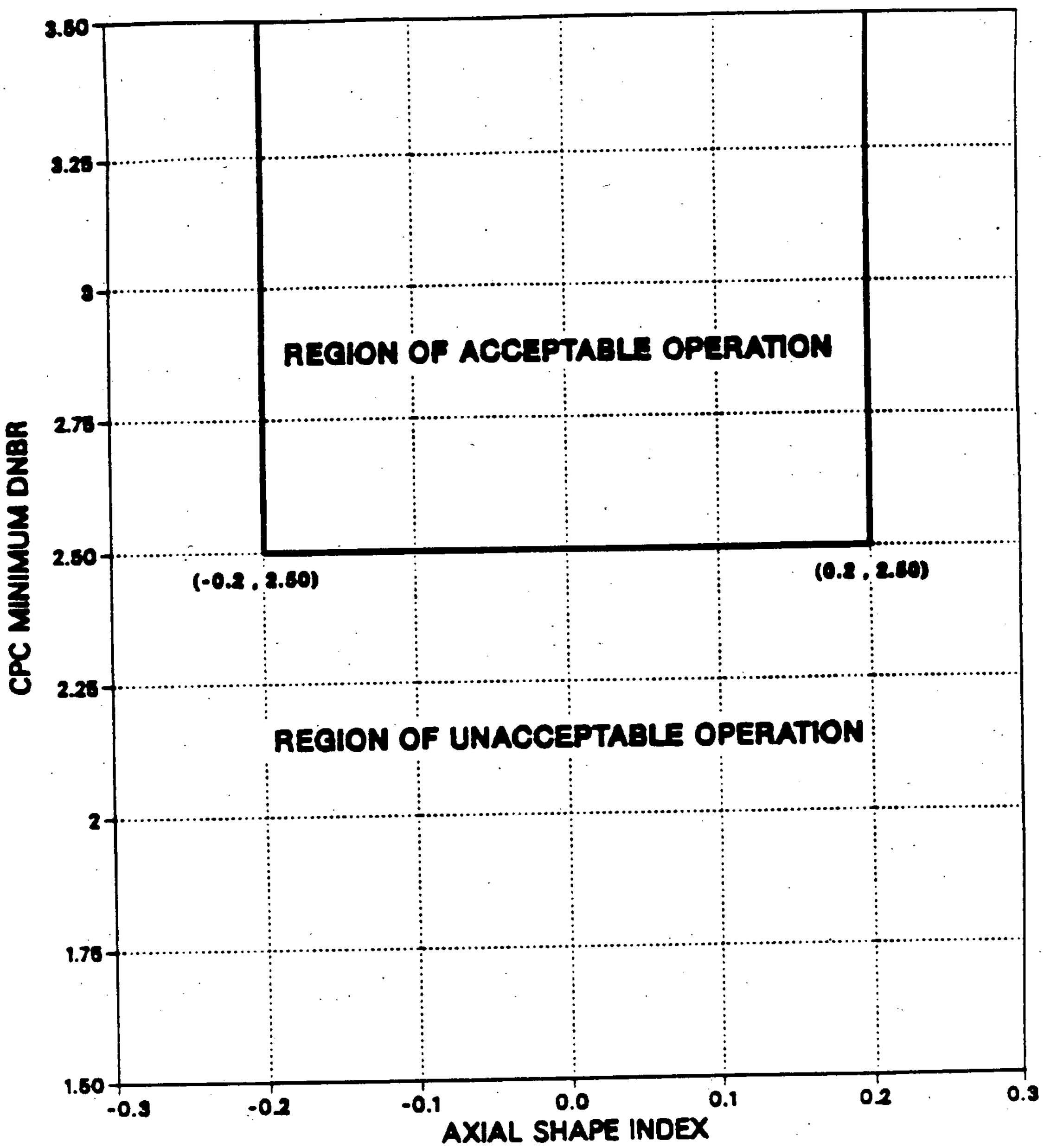

FIGURE 3.2-2 DNBR OPERATING LIMIT BASED ON CORE PROTECTION CALCULATOR
- COLSS OUT OF SERVICE
- BOTH CEACS INOPERABLE

DELETED INTENTIONALLY

DELETED INTENTIONALLY

DELETED INTENTIONALLY

ACTION STATEMENTS

2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 2 are satisfied.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 6 -

a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEA's in its group. After 7 days, operation may continue provided that ACTION 6.b is met.*

b. With both CEACs inoperable, operation may continue provided that:*

1. Within 1 hour the DNBR margin required by Specification 3.2.4.b (COLSS in service) or Specification 3.2.4.d (COLSS out of service) is satisfied.

*Note: Requirements for CEA position indication given in Technical Specification 3.1.3.2.

TABLE NOTATION

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPC's is set to indicate that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEA's are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEA's are aligned within 7 inches (indicated position) of all other CEA's in its group.

ACTION 7 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 7A - 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.

DELETED INTENTIONALLY

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.

3/4.2.4 DNBR MARGIN

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. The COLSS calculation of core power operating limit based on the minimum DNBR limit includes appropriate penalty factors which provide, with a 95/95 probability/confidence level, that the core power limit calculated by COLSS (based on the minimum DNBR limit) is conservative with respect to the actual core power limit. The penalty factors are determined from the uncertainties associated with planar radial peaking measurement, engineering design factors, state parameter measurement, software algorithm modelling, computer processing, rod bow and core power measurement.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. In the event that the COLSS is not being used, the DNBR margin can be maintained by monitoring with any operable CPC channel so that the DNBR remains above the predetermined limit as a function of Axial Shape Index. The above listed uncertainty penalty factors are also included in the CPCs, which assume a minimum of 20% of RATED THERMAL POWER. The 20% RATED THERMAL POWER threshhold is due to the excore neutron flux detector system being less accurate below 20% core power. Core noise level at low power is too large to obtain usable detector readings. The additional uncertainty terms taken into account in the CPCs for transient protection are removed from Figures 3.2-1 and 3.2-2 since the curves are intended to monitor the LCO only during steady state operation.

DNBR Margin (continued)

A DNBR penalty factor has been included in the COLSS and CPC DNBR calculation to accommodate the effects of rod bow. The amount of rod bow in each fuel assembly is dependent upon the average burnup experienced by that assembly. Fuel assemblies that incur higher average burnup will experience a greater magnitude of rod bow. Conversely, lower burnup assemblies will experience less rod bow. In design calculations, the penalty for each batch required to compensate for rod bow is determined from a batch's maximum average assembly burnup applied to the batch's maximum integrated planar-radial power peak. A single net penalty for COLSS and CPC is then determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

3/4.2.5 RCS FLOW RATE

This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

This specification is provided to ensure that the actual value of reactor coolant cold leg temperature is maintained within the range of values used in the safety analyses.

2.4.2.7 AXIAL SHAPE INDEX

This specification is provided to ensure that the actual value of AXIAL SHAPE INDEX is maintained within the range of values used in the safety analyses.

3/4.2.8 PRESSURIZER PRESSURE

This specification is provided to ensure that the actual value of pressurizer pressure is maintained within the range of values used in the safety analyses.

DESCRIPTION OF PROPOSED CHANGES NPF-10/15-202 AND SAFETY ANALYSIS

This is a request to revise Technical Specification 3/4.1.1.3, "Moderator Temperature Coefficient".

Existing Technical Specifications

Unit 2: See Attachment A
Unit 3: See Attachment C

Proposed Technical Specifications

Unit 2: See Attachment B
Unit 3: See Attachment D

Description

The proposed change revises Technical Specification 3/4.1.1.3, "Moderator Temperature Coefficient". Technical Specification 3/4.1.1.3 imposes limitations on moderator temperature coefficient (MTC) to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle will be performed to confirm the MTC value since this coefficient changes slowly due principally to the reduction in reactor coolant system (RCS) boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

Technical Specification 3/4.1.1.3 currently states that the moderator temperature coefficient shall be less negative than -2.5×10^{-4} delta k/k/^{o}F at rated thermal power. The proposed change will state that the moderator temperature coefficient shall be less negative than -3.3×10^{-4} delta k/k/^{o}F. This change is required to reflect the use of more highly enriched fuel in uranium-235 beginning in Cycle 3 and the increased burnup to which this fuel will be subjected by end of cycle.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

Response: No

The proposed negative moderator temperature coefficient change was incorporated as an assumption in the Cycle 3 transient analysis and thus remains accounted for in an equally conservative manner as before. The events most affected by the change are those characterized by a decrease in primary temperature. Details of these analyses demonstrate that although the proposed change may be perceived to slightly increase in some way the consequences of an accident, the results of the change are clearly within all acceptable criteria with respect to the system or component of concern as specified in the Standard Review Plan, Section 4.3.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   No change to operating procedures is involved. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed negative moderator temperature coefficient change was incorporated as an assumption in the Cycle 3 transient analysis and thus remains accounted for in equally conservative manner as before. The events most affected by the change are those characterized by a decrease in primary temperature. Details of these analyses are presented in the Reload Analyses Report for Cycle 3. These analyses demonstrate that although the proposed change may be perceived to slightly increase in some way the consequences of an accident, the results of the change are clearly within all acceptable criteria with respect to the system or component of concern as specified in the Standard Review Plan, Section 4.3.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (iii) relates to a change resulting from a nuclear reactor reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that the NRC has previously found such methods acceptable. The proposed change is similar to Example (iii) in that the technical specification on the moderator temperature coefficient will reflect the use of more highly enriched fuel that will be exposed to increased burnup by end of cycle. This change is not significant and does not make changes in analytical methods used to demonstrate conformance with the technical specifications and regulations, and the NRC has previously found such methods acceptable.

Safety and Significant Hazards Determination

Based on the above Safety Analysis, it is concluded that: (1) there is a reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:4755F

ATTACHMENT A

Existing Technical Specifications, Unit 2

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is ≤ 70% of RATED THERMAL POWER, or

Less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -2.5×10^{-4} delta k/k/°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 at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 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.

4.1.1.3.2 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 of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

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ATTACHMENT B

Proposed Technical Specifications, Unit 2

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or

Less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -3.3×10^{-4} delta k/k/°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 at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 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.

4.1.1.3.2 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 of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

ATTACHMENT C

Existing Technical Specifications, Unit 3

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, ~~and~~

b. Less negative than -2.5×10^{-4} delta k/k/°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 at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 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.

4.1.1.3.2 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 of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

ATTACHMENT D

Proposed Technical Specifications, Unit 3

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} delta k/k/°F whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER, or less positive than 0.0 delta k/k/°F whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

b. Less negative than -3.3×10^{-4} delta k/k/°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 at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 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.

4.1.1.3.2 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 of reaching 40 EFPD core burnup.

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup.

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

#See Special Test Exception 3.10.2.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-203

This is a request to revise Technical Specification 3/4.3.1, "Reactor Protective Instrumentation," and Technical Specification 3/4.2.4, "DNBR Margin."

Existing Technical Specifications

Unit 2: See Attachment A

Unit 3: See Attachment C

Proposed Technical Specifications

Unit 2: See Attachment B

Unit 3: See Attachment D

Description

The proposed changes revise Technical Specification 3/4.3.1, "Reactor Protective Instrumentation," and Technical Specification 3/4.2.4, "DNBR Margin." Technical Specification 3/4.3.1 requires that the Reactor Protective Instrumentation System (RPIS) be operable and defines the number and type of RPIS channels required, their setpoints and their response times and periodic testing requirements to assure operability. This technical specification defines maximum reactor protection instrumentation response times in order to verify that the maximum response time for RPIS assumed in the Final Safety Analysis Report (FSAR) are not exceeded. Technical Specification 3/4.2.4 requires that the departure from nucleate boiling ratio (DNBR) be maintained by operating within the region of acceptable operation as indicated by either the Core Operating Limit Supervisory System (COLSS) or the Core Protection Calculator (CPC).

The proposed changes consists of the following parts:

a. The proposed change revises Table 3.3-2, "Reactor Protective Instrumentation Response Times," Table 3.3-2a, "Increases in BERR Constants," and Table 3.3-2b, "DNBR LCO Power Operating Limit Adjustments," of Technical Specification 3/4.3.1. Specifically Item 10, "DNBR-Low," specifies a response time of 0.68 second for RCS Hot Leg and Cold Leg Temperature. The table notes that these response times are based on a resistance temperature detector (RTD) response time of less than or equal to 13.0 seconds. For Cycles 1 and 2, the accident analysis used an initial RTD response time of 6 seconds. Table 3.3-2a and 3.3-2b require adjustments to the CPC addressable constants and the reduction in the DNBR power operating limit in the COLSS to compensate for measured RTD response times greater than 6 seconds. For Cycle 3, the accident analysis use an RTD response time of 8.0 seconds. The proposed change revises the RTD maximum response time from 13.0 seconds to 8.0 seconds in Note (##) of Table 3.3-2, and deletes Tables 3.3-2a and 3.3-2b, since adjustments are no longer necessary for response times greater than 6 seconds. For consistency, the references to Table 3.3-2b are deleted from Surveillance Requirement 4.2.4.2 of Specification 3/4.2.4.

b. The proposed change also revises Note (#) appended to Item 10(e), "Primary Coolant Pump Shaft Speed." It presently states that the response time shall be measured from the onset of 2 out of 4 Reactor Coolant Pump coastdown. This note would be revised to clarify that the response time is measured using simulated pump coastdown.

One of the ways to simulate the reactor coolant pump (RCP) coastdown is using the response time test box. The fly-disk on the RCP generates the pulses which are sent to the CPC. The CPC counts the pulses to determine the frequency, which represents the RCP speed. Appropriate initial conditions are entered in the CPC. The pump coastdown is then simulated by changing the frequency using the response time test box that would represent the pump speed at the end of the transient marking the point at which the CPC generates a trip. Therefore, response time measurement is done using the simulation rather than from the onset of a 2 out of 4 RCP coastdown.

<u>Safety Analysis</u>

The proposed change discussed above shall be deemed to constitute a significant hazards consideration if there is a positive finding in any of the following areas.

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   Since the proposed changes in part (a) to revise the RTD response time does not alter how the CPC responds to design basis events, the CPC will continue to perform its functions as before. Also, the proposed change in part (b) to measure the response time using a simulated pump coastdown does not affect the CPC's functions to provide the protection. Therefore, the operation of the facility in accordance with the proposed amendment will not increase the probability or consequences of an accident previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Since operation of the facility in accordance with the proposed changes will not change, there is no new or different kind of accident from any accident previously evaluated that could occur.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

   Response: No

The CPC responds to design basis transient events based in part on information from the RTD's. The proposed changes specify a more restrictive allowed maximum RTD response time and evaluation of the low DNBR trip functions in the proposed CPC software. This allows the CPC to continue to provide the same degree of protection for the transients previously evaluated in the safety analysis report. Also, there is no change in how CPC responds to design basis events. Furthermore, the proposed change in part (b) to measure the response time using a simulated pump coastdown does not affect the response time measurement or protection provided by the RPIS. Therefore, operation of the facility in accordance with these proposed amendments will not involve a reduction in a margin of safety.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (ii) relates to changes which may constitute an additional limitation, restriction, or control not presently included in the Technical Specifications. Example (i) relates to a purely administrative change to the technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature. The proposed change in part (a) would revise the RTD response time from 13.0 seconds to 8.0 seconds. This change is similar to Example (ii) in that the proposed RTD response time is more restrictive than the current one. The proposed change described in part (b) above is similar to example (i) in that it more accurately specifies the method for response time testing.

Safety and Significant Hazards Determination

Based on the above safety analysis, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change: and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PS:4758F

ATTACHMENT A

(Existing Technical Specification)

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 13.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. Adjustments to the CPC addressable constants in Table 3.3-2a and reductions in the DNBR Power Operating Limit in Table 3.3-2b shall be made to accommodate measured values of RTD time constants.

TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD Delay Time τ	BERR0 Increase %		BERR2 Increase %		BERR4 Increase %	
	Cycle 1	Cycle 2	Cycle 1	Cycle 2	Cycle 1	Cycle 2
$\tau \leq 6.0$ sec.	0.0	0.0	0.0	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	0.0	2.0	3.5	5.0	3.0	0.0
8.0 sec. $< \tau \leq$ 10.0 sec.	3.5	5.0	4.0	8.5	9.0	3.0
10.0 sec. $< \tau \leq$ 13.0 sec.	10.5	9.0	5.5	12.0	17.0	6.0

NOTE: BERR term increases are not cumulative, i.e., if the values of the BERR terms are currently 10.0, then for an RTD delay time of > 6.0 to ≤ 8.0 sec., in Cycle 1: BERR0 = 10.0 + 0.0 = 10.0; BERR2 = 10.0 + 3.5 = 13.5; and, BERR4 = 10.0 + 3.0 = 13.0. For RTD delay times of > 8.0 to ≤ 10.0 sec., in Cycle 1: BERR0 = 10.0 + 3.5 = 13.5; BERR2 = 10.0 + 4.0 = 14.0; and BERR4 = 10.0 + 9.0 = 19.0. Computed values in this paragraph are examples only.

NOTE: In Cycle 1 only, when any of the above increases are applied to the BERR terms for any CPC channel, the COLSS constant EPOL2 is reduced by 0.04. This applies for Cycle 1 only.

TABLE 3.3-2b

DNBR LCO POWER OPERATING LIMIT ADJUSTMENTS

RTD Delay Time (sec)	Adjustment to EPOL1,[1] COLSS In Service (% power)	Adjustment to BERR2,[1,2] COLSS Out of Service (% power) Cycle 1	Cycle 2
$\tau \leq 6.0$ sec.	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	-4.0	+4.0	+5.0
8.0 sec. $< \tau \leq$ 10.0 sec.	-5.0	+5.0	+8.5
10.0 sec $< \tau \leq$ 13.0 sec.	-7.0	+7.0	+12.0

NOTES:

1. Adjustments are not cumulative: i.e., if τ increases from 7.0 seconds to 9.0 seconds, EPOL1 is reduced by 5.0 from its original value, not 4.0 + 5.0 = 9.0 from its original value.

2. If COLSS is out-of-service, these adjustments are to be used in place of, not in addition to, the increases required by Table 3.3-2a and the limit in Figure 3.2-2 or 3.2-3, as applicable, must be maintained for all operable CPC channels.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable and the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

ATTACHMENT B

(Proposed Technical Specification)

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured using a simulated Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 8.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

DELETED

DELETED

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

ATTACHMENT C

(Existing Technical Specification)

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 13.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature. Adjustments to the CPC addressable constants in Table 3.3-2a and reductions in the DNBR Power Operating Limit in Table 3.3-2b shall be made to accommodate measured values of RTD time constants.

TABLE 3.3-2a

INCREASES IN BERR0, BERR2, AND BERR4 VERSUS RTD DELAY TIMES

RTD Delay Time τ	BERR0 Increase %		BERR2 Increase %		BERR4 Increase %	
	Cycle 1	Cycle 2	Cycle 1	Cycle 2	Cycle 1	Cycle 2
$\tau \leq 6.0$ sec.	0.0	0.0	0.0	0.0	0.0	0.0
6.0 sec. $< \tau \leq$ 8.0 sec.	0.0	2.0	3.5	5.0	3.0	0.0
8.0 sec. $< \tau \leq$ 10.0 sec.	3.5	5.0	4.0	8.5	9.0	3.0
10.0 sec. $< \tau \leq$ 13.0 sec.	10.5	9.0	5.5	12.0	17.0	6.0

NOTE: BERR term increases are not cumulative, i.e., if the values of the BERR terms are currently 10.0, then for an RTD delay time of > 6.0 to ≤ 8.0 sec., in Cycle 1: BERR0 = 10.0 + 0.0 = 10.0; BERR2 = 10.0 + 3.5 = 13.5; and, BERR4 = 10.0 + 3.0 = 13.0. For RTD delay times of > 8.0 to ≤ 10.0 sec., in Cycle 1: BERR0 = 10.0 + 3.5 = 13.5; BERR2 = 10.0 + 4.0 = 14.0; and BERR4 = 10.0 + 9.0 = 19.0. Computed values in this paragraph are examples only.

NOTE: In Cycle 1 only, when any of the above increases are applied to the BERR terms for any CPC channel, the COLSS constant EPOL2 is reduced by 0.04. This applies for Cycle 1 only.

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TABLE 3.3-2b

DNBR LCO POWER OPERATING LIMIT ADJUSTMENTS

RTD Delay Time (sec)	Adjustment to EPOL1,[1] COLSS In Service (% power)	Adjustment to BERR2,[1,2] COLSS Out of Service (% power)	
		Cycle 1	Cycle 2
$\tau \leq$ 6.0 sec.	0.0	0.0	0.0
6.0 sec. < $\tau \leq$ 8.0 sec.	-4.0	+4.0	+5.0
8.0 sec. < $\tau \leq$ 10.0 sec.	-5.0	+5.0	+8.5
10.0 sec. < $\tau \leq$ 13.0 sec.	-7.0	+7.0	+12.0

NOTES:

1. Adjustments are not cumulative: i.e., if τ increases from 7.0 seconds to 9.0 seconds, EPOL1 is reduced by 5.0 from its <u>original</u> value, <u>not</u> 4.0 + 5.0 = 9.0 from its original value.

2. If COLSS is out-of-service, these adjustments are to be used in place of, not in addition to, the increases required by Table 3.3-2a and the limit in Figure 3.2-2 or 3.2-3, as applicable, must be maintained for all operable CPC channels.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable and the conditions of Table 3.3-2b are satisfied.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

ATTACHMENT D

(Proposed Technical Specification)

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured using a simulated Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 8.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

DELETED

DELETED

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-1, 3.2-2, or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to restore the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-2 or 3.2-3, whichever is applicable.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-204

This is a request to revise Technical Specification 2.2.1, "Reactor Trip Setpoints," and its associated bases.

Existing Specification

Unit 2: See Attachment A

Unit 3: See Attachment C

Proposed Specifications

Unit 2: See Attachment B

Unit 3: See Attachment D

Description

The proposed change revises Technical Specification 2.2.1, "Reactor Trip Setpoints," and its associated bases. Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," requires that the setpoints for trip values of the Reactor Protective System (RPS) be set at specified values and kept within a specified allowable value range. The Local Power Density - High Trip Setpoint of Table 2.2-1 specifies the required trip setpoint to prevent the peak linear heat rate from exceeding its safety limit for transients and anticipated operational occurrences and to mitigate the consequences of accidents. The proposed change revises the value of the Local Power Density - High Trip Setpoint and Allowable Value of LSSS 2.2.1, Table 2.2-1. Specifically, Table 2.2-1, Functional Unit 9, requires that both the Trip Setpoint and Allowable Value for the Local Power Density - High trip be 19.95 kw/ft. The proposed change increases both the Trip Setpoint and Allowable Value to 21.0 kw/ft, and reflects this revised setpoint in the Bases. This trip setpoint is a generic value with which the Core Protection Calculator System (CPCS) acts to prevent the Safety Limit from being exceeded during anticipated operating occurrences. Previously the trip setpoint incorporated an adjustment for dynamic effects that will now be accounted for elsewhere in the CPCS algorithms. Any changes to this adjustment on a cycle dependent basis can be incorporated in the addressable constants. Effectively, the CPCS local power density protection is not being changed.

General Design Criterion 10 - Reactor Design, requires that the reactor core and associated coolant, control and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The specified trip settings result in confidence that the specified acceptable fuel design limits will not be exceeded during normal operation or as the result of anticipated operational occurrences.

Safety Analysis

The proposed change described above shall be deemed to involve a significant hazards consideration if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   The proposed change reflect revisions in setpoints and CPC's software for the cycle 3 reload analysis which ensure that specified acceptable fuel design limits are not exceeded for anticipated operational occurrences.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   Since the operation of the facility in accordance with the proposed amendment will not change, there is no new or different kind of accident from any accident previously evaluated that could occur.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   Since the CPC's will continue to provide the same degree of protection for anticipated operational occurrences and preserves the NRC acceptance criteria for accidents, operation of the facility in accordance with this proposed amendment will not involve a reduction in a margin of safety.

The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered least likely to involve significant hazards considerations. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan: for example a change resulting from the application of a small refinement of a previously used calculational model or design method.

SRP Section 7.2, "Reactor Trip System", requires that the reactor protection system automatically initiates a reactor trip to assure that specified acceptable fuel design limits are not exceeded. The proposed change is similar to example (vi). Although the increased linear power density (LPD) trip setpoint may be perceived to reduce in some way a margin of safety, adjustments for dynamic effects which were previously included in the trip setpoint are now accounted for elsewhere in the CPC algorithms. The net effect is that the CPC's with the revised setpoint, will continue to initiate a reactor trip to assure that specific acceptable fuel design limits are not exceeded. Therefore the proposed change satisfies SRP section 7.2 acceptance criteria and is similar to example (vi).

Safety and Significant Hazards Consideration Determining

Based on the Safety Evaluation, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change: and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

(Existing Technical Specification)

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 19.95 kw/ft	≤ 19.95 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The Reactor Coolant System components are designed to Section III, 1971 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.31 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

ATTACHMENT B

(Proposed Technical Specification)

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 21.0 kw/ft	≤ 21.0 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The Reactor Coolant System components are designed to Section III, 1971 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.31 and 21.0 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

ATTACHMENT C

(Existing Technical Specification)

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	≤ 110.0% of RATED THERMAL POWER	≤ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia	≥ 1763 psia
6. Containment Pressure - High	≤ 2.95 psig	≤ 3.14 psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia	≥ 711 psia
8. Steam Generator Level - Low	≥ 25% (4)	≥ 24.23% (4)
9. Local Power Density - High (5)	≤ 19.95 kw/ft	≤ 19.95 kw/ft
10. DNBR - Low	≥ 1.31 (5)	≥ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	≤ 0.22 psid/sec (6)(8)	≤ 0.231 psid/sec (6)(8)
b) Floor	≥ 13.2 psid (6)(8)	≥ 12.1 psid (6)(8)
c) Step	≤ 6.82 psid (6)(8)	≤ 7.231 psid (6)(8)
12. Steam Generator Level - High	≤ 90% (4)	≤ 90.74% (4)
13. Seismic - High	≤ 0.48/0.60 (7)	≤ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The Reactor Coolant System components are designed to Section III, 1971 Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

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The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.31 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

ATTACHMENT D

(Proposed Technical Specification)

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 21.0 kw/ft	$\leq$ 21.0 kw/ft
10. DNBR - Low	$\geq$ 1.31 (5)	$\geq$ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.22 psid/sec (6)(8)	$\leq$ 0.231 psid/sec (6)(8)
b) Floor	$\geq$ 13.2 psid (6)(8)	$\geq$ 12.1 psid (6)(8)
c) Step	$\leq$ 6.82 psid (6)(8)	$\leq$ 7.231 psid (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

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To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-147(S)-P, "Functional Design Specification for a Core Protection Calculator," January, 1981; CEN-148(S)-P, "Functional Design Specification for a Control Element Assembly Calculator," January, 1981; CEN-149(S)-P "CPC/CEAC Data Base Document", January, 1981, and CEN-175(S)-P "SONGS 2 Cycle 1 CPC and CEAC Data Base Document", August, 1981.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Description of Proposed Change NPF-10/15-206
and Safety Analysis

This is a request to revise Technical Specifications 2.2.2, "Core Protection Calculator Addressable Constants", 6.5.1.6, "Onsite Review Committee", 6.8.1 "Procedures and Programs", and Bases 3/4.3.1, "Reactor Protective Instrumentation".

Existing Technical Specifications

Unit 2 - See Attachment A
Unit 3 - See Attachment B

Proposed Technical Specifications

Unit 2 - See Attachment C
Unit 3 - See Attachment D

Description

The proposed change revises Technical Specifications 2.2.2, "Core Protection Calculator Addressable Constants", 6.5.1.6, "Onsite Review Committee", 6.8.1, "Procedures and Programs," and the Bases for Specifications 3/4.3.1 and 3/4.3.2, "Reactor Protective and Engineered Safety Features Actuation System Instrumentation". These specifications relate to core protection calculator (CPC) addressable constants and define allowable ranges for certain constants and administrative controls on changes to addressable constants values.

The CPC's are digital computers which form part of the reactor protection system which monitors various plant parameters and automatically generate a reactor trip when specified conditions are exceeded. The CPC's monitor various reactor parameters and calculate departure from nucleate boiling ratio (DNBR) and local power density (LPD). If DNBR and LPD are calculated to exceed the DNBR and LPD trip setpoints, the CPC's automatically initiate a reactor trip.

The CPC addressable constants provide a mechanism to calibrate CPC calculations to account for physics tests measurements, uncertainties, sensor operability status, flow and power measurements, and changes in core design from cycle to cycle. Addressable constants are categorized as either Type I or Type II. Type I constants are calibrating constants, sensor operability status flags and pre-trip alarm setpoints which potentially could change frequently during a fuel cycle. Type I addressable constant values are entered using the CPC operator module.

Type II addressable constants are related to measured physics test parameters, uncertanties, allowances, adjustments and values determined during startup tests following each fuel loading, and are not expected to change during cycle operation. Values for Type II addressable constants are typically entered via diskettes but also can be entered via the operator module.

The CPC software includes range limits on addressable constants. These range limits prevent gross errors from being made during addressable constant entry. However, the range limit values are not related to the safety analysis; that is having addressable constant values within the software limits does not in itself guarantee conservative CPC operation.

Technical Specification 2.2.2 "CPC Addressable Constants", requires that CPC addressable constants be in accordance with Table 2.2-2. Table 2.2-2 is a list of Type I and Type II addressable constants. Ranges of allowable values are included for Type I Addressable Constants. The specified range of values are more restrictive than the software limits but are not related to the safety analysis and in themselves do not guarantee conservative CPC operation. Actual values for Type I addressable constants result from other technical specification surveillance and action requirements (e.g., 3/4.3.1 Reactor Protective Instrumentation, 3/4.2.3 Azimathal Power Tilt).

Technical Specification 6.5.1.6, "Onsite Review Committee", requires that the Onsite Review Committee (OSRC) review and approve the entry of addressable constant values outside the allowable range of Table 2.2-2.

Technical Specification 6.8.1, "Procedures and Programs", requires that written procedures be established to cover modification of CPC addressable constant values and requires prior OSRC approval of entry of CPC addressable constant values based on information obtained via the Plant Computer-CPC data link.

The proposed change deletes (1) TS 2.2.2 and associated Table 2.2-2, (2) the TS 6.5.1.6 requirement for OSRC review and approval of the entry of addressable constant values outside the allowable range of Table 2.2-2, and (3) the TS 6.8.1 requirement of OSRC approval of entry of CPC addressable constant values based on information obtained via the Plant Computer-CPC data link. In addition, the proposed change would revise Bases Section 3/4.3.1, "Reactor Protective Instrumentation", to identify (1) the general function of CPC addressable constants, (2) that the potential for inadvertent misloading of addressable constants is minimized by administrative controls, (3) that modifications to CPC software will be made in accordance with an NRC approved procedure, and (4) that CPC software modifications which involve either an unreviewed safety question, technical specifications changes, or new methodology previously not reviewed by the NRC will require NRC approval prior to implementation.

Safety Analysis

The proposed change discussed above shall be deemed to involve significant hazards considerations if there is a positive finding in any of the following areas:

1. Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change will remove TS range limits on Type I CPC addressable constants and remove the requirement for OSRC approval for loading addressable constants using the plant computer data link. The actual values of Type I addressable constants are governed by existing TS surveillance requirements (e.g., 3/4.3.1) and action statements (e.g., 3/4.2.3). Type I addressable constant values derived in accordance with the TS requirements but outside the allowable range of Table 2.2-2 currently can be entered with OSRC approval (TS 6.5.1.6). Additionally, the allowable ranges for Type I addressable constants are not related to the safety analysis and use of values in the specified ranges does not in itself guarantee conservative CPC operation. However, adjusting the values of the Type I addressable constants in accordance with reactor protective instrumentation surveillances and actions does ensure conservative CPC operation.

The proposed change would not require prior OSRC approval for use of the Plant Computer data link in the generation of addressable constant values. Addressable constant values can be generated without the use of the data link and without OSRC approval. Currently data used in the generation of addressable constants can be manually obtained from the CPC's. Manual recording of CPC data is more error prone than using the data link. Loading of addressable constants will continue to be controlled by written procedures established in accordance with TS 6.8.1.

The proposed change affects the administrative controls for values of addressable constants and does not affect CPC function. Therefore, the proposed change does not affect the way the CPC's would respond in accidents and transients previously evaluated.

2. Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not alter the configuration of the facility or its operation. Therefore the proposed change does not create the possibility of a new or different kind of accident.

3. Will operation of the facility in accordance with this proposed amendment involve a significant reduction in a margin of safety?

   Response: No

   The proposed change does not affect operation of the CPC's. The CPC's will continue to function as before in accidents and transients. Therefore the proposed change does not reduce a margin of safety.

The Commission has provided guidance concerning the application of standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered not likely to involve significant hazards considerations. Example (i) relates to a purely administrative change to the technical specifications: for example a change to achieve consistency throughout the technical specifications, correction of an error or a change in nomenclature. Example (vi) relates to a change which either may result in some increase in the probability or consequences of a previously analyzed accident or may in some way reduce a margin of safety but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plant (SRP).

The proposed change deletes TS 2.2.2 (and Table 2.2-2) and the TS 6.5.1.6 requirement for OSRC review and approval for the use of addressable constant values outside the allowable ranges stated in TS 2.2.2. The proposed change amplifies Bases Section 3/4.3.1 in regards to the function and control of changes to addressable constants.

The allowable range of values in TS 2.2.2 are not related to the safety analysis and the use of values within these ranges does not ensure conservative CPC operation. Actual values to be used for addressable constants result from surveillance requirements for reactor protective instrumentation (TS 3/4.3.1) and action requirements (e.g., TS 3/4.6.1 Action 6c which requires adjustments to addressable constant values when certain equipment is out of service). There have been occasions where satisfaction of the surveillance requirements have resulted in addressable constant values outside the allowable range, a conflict in existing TS requirements. Because of this conflict and because other TS's control the actual values of addressable constants, the proposed deletions are editorial and correct an existing conflict. Therefore the proposed change is similar to example (i).

The proposed change to TS 6.8.1 deleting the requirements for prior OSRC approval of the use of the plant computer - CPC data link may be perceived to in some way decrease a safety margin. It should be noted that values for CPC addressable constants can be generated from CPC data manually collected from the CPC's without OSRC approval and that manual collection of data would be more prone to errors than use of the data link. The proposed change does not affect the functioning of the CPC's as part of the reactor protection system but merely revises administrative control of how addressable constant values are generated. Thus with the proposed change the CPC's will continue to satisfy the SRP Section 7.2, "Reactor Trip System", requirement for the reactor protection system to automatically initiate a reactor trip to assure that specified acceptable fuel design limits are not exceeded. Therefore this change is similar to example (vi).

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it 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. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PS:4949F

Attachment "A"

Unit 2 Existing Specifications

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	$\leq$1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	$\geq$1.02
64	TPC	Thermal power calibration constant	$\geq$0.90
65	KCAL	Neutron flux power calibration constant	$\geq$0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted
98	TCREF	Reference Cold Leg Temperature	520°F $\leq$ TCREF $\leq$ 580°
104	PCALIB	Calorimetric Power	$\leq$102.0

MAR 0 1 1985

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERR0	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient
103	RPCLIM	Reactor Power Cutback Time Limit

MAR 01 1985

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

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

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

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

MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the OSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Onsite Review Committee shall be responsible for:

a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

b. Review of events requiring 24-hour written notification to the Commission.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

f. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15 Rev. 1, February 1979.

j. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Onsite Review Committee.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Manager, Operations (2) the Manager, Technical (3) the Manager, Maintenance, (4) the Deputy Station Manager, or (5) the Manager, Health Physics as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only) and the liquid radwaste system (post-accident sampling return piping only). The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

*See Specification 6.13.1

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Attachment "B"

Unit 2 Proposed Specification

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

Any modifications which are made to the core protection calculator software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revisions 2 and Supplement 1-P, Revision 01 or another NRC approved procedure on CPC software modifications.

CPC modifications which result in a) an unreviewed safety questions, b) a Technical Specification change, or c) methodology not previously approved by the NRC, including additions or deletions to addressable constants will require NRC approval prior to implementation.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

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

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

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

MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the OSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Onsite Review Committee shall be responsible for:

a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

b. Review of events requiring 24-hour written notification to the Commission.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15 Rev. 1, February 1979.

j. Modification of Core Protection Calculator (CPC) Addressable Constants.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Manager, Operations (2) the Manager, Technical (3) the Manager, Maintenance, (4) the Deputy Station Manager, or (5) the Manager, Health Physics as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. <u>Primary Coolant Sources Outside Containment</u>

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only) and the liquid radwaste system (post-accident sampling return piping only). The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

*See Specification 6.13.1

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Attachment "C"

Unit 3 Existing Specifications

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

2.2.2 Core Protection Addressable Constants shall be in accordance with Table 2.2-2.

APPLICABILITY: As shown for Core Protection Calculators in Table 3.3-1.

ACTION:

With a Core Protection Calculator Addressable Constant found to be non-conservative, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status.

NOV 15 1982

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	$\leq$1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	$\geq$1.02
64	TPC	Thermal power calibration constant	$\geq$0.90
65	KCAL	Neutron flux power calibration constant	$\geq$0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

NOV 1 5 1982

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
68	BERRO	Thermal power uncertainty bias
69	BERR1	Power uncertainty factor used in DNBR calculation
70	BERR2	Power uncertainty bias used in DNBR calculation
71	BERR3	Power uncertainty factor used in local power density calculation
72	BERR4	Power uncertainty bias used in local power density calculation
73	EOL	End of life flag
74	ARM1	Multiplier for planar radial peaking factor
75	ARM2	Multiplier for planar radial peaking factor
76	ARM3	Multiplier for planar radial peaking factor
77	ARM4	Multiplier for planar radial peaking factor
78	ARM5	Multiplier for planar radial peaking factor
79	ARM6	Multiplier for planar radial peaking factor
80	ARM7	Multiplier for planar radial peaking factor
81	SC11	Shape annealing correction factor
82	SC12	Shape annealing correction factor
83	SC13	Shape annealing correction factor
84	SC21	Shape annealing correction factor
85	SC22	Shape annealing correction factor
86	SC23	Shape annealing correction factor
87	SC31	Shape annealing correction factor
88	SC32	Shape annealing correction factor

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION
89	SC33	Shape annealing correction factor
90	PFMLTD	DNBR penalty factor correction multiplier
91	PFMLTL	LPD penalty factor correction multiplier
92	ASM2	Multiplier for CEA shadowing factor
93	ASM3	Multiplier for CEA shadowing factor
94	ASM4	Multiplier for CEA shadowing factor
95	ASM5	Multiplier for CEA shadowing factor
96	ASM6	Multiplier for CEA shadowing factor
97	ASM7	Multiplier for CEA shadowing factor
98	CORR1	Temperature shadowing correction factor multiplier
99	BPPCC1	Boundary point power correlation coefficient
100	BPPCC2	Boundary point power correlation coefficient
101	BPPCC3	Boundary point power correlation coefficient
102	BPPCC4	Boundary point power correlation coefficient

NOV 15 1982

2.2.2 CPC ADDRESSABLE CONSTANTS

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flow rate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPC's is unlikely.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

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

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

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

NOV 1 5 1982

MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the OSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Onsite Review Committee shall be responsible for:

a. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

b. Review of events requiring 24-hour written notification to the Commission.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

f. Review and approval of using and entering values of CPC addressable constants outside the allowable range of Table 2.2-2.

AUTHORITY

6.5.1.7 The Onsite Review Committee (OSRC) shall:

a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unreviewed safety questions.

b. Provide written notification within 24 hours to the Manager of Nuclear Operations and NSG of disagreement between the OSRC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

NOV 1 5 1982

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the OSRC and submitted to the NSG and the Manager of Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager of Nuclear Operations and the NSG Chairman shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the OSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Manager of Nuclear Operations and the NSG within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

NOV 15 1982

g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15 Rev. 1, February 1979.

j. Modification of Core Protection Calculator (CPC) Addressable Constants.

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Onsite Review Committee.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only) and the liquid radwaste system (post-accident sampling return piping only). The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

*See Specification 6.13.1

NOV 15 1982

Attachment "D"

Unit 3 Proposed Specifications

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

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

When a protection channel of a given process variable becomes inoperable, the inoperable channel may be placed in bypass until the next Onsite Review Committee meeting at which time the Onsite Review Committee will review and document their judgment concerning prolonged operation in bypass, channel trip, and/or repair. The goal shall be to return the inoperable channel to service as soon as practicable but in no case later than during the next COLD SHUTDOWN. This approach to bypass/trip in four channel protection systems is consistent with the applicable criteria of IEEE Standards 279, 323, 344 and 384.

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Any modifications which are made to the core protection calculator software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revisions 2 and Supplement 1-P, Revision 01 or another NRC approved procedure on CPC software modifications.

CPC modifications which result in a) an unreviewed safety questions, b) a Technical Specification change, or c) methodology not previously approved by the NRC, including additions or deletions to addressable constants will require NRC approval prior to implementation.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The redundancy and design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEAC's becomes inoperable. If one CEAC is in test or inoperable, verification of CEAC position is performed at least every 4 hours. If the second CEAC fails, the CPC's will use DNBR and LPD penalty factors, which restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded a reactor trip will occur.

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MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

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RESPONSIBILITIES

6.5.1.6 The Onsite Review Committee shall be responsible for:

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b. Review of events requiring 24-hour written notification to the Commission.

c. Review of unit operations to detect potential nuclear safety hazards.

d. Performance of special reviews, investigations or analyses and reports thereon as requested by the Station Manager or the NSG.

e. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last OSRC meeting.

AUTHORITY

6.5.1.7 The Onsite Review Committee (OSRC) shall:

a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unreviewed safety questions.

b. Provide written notification within 24 hours to the Manager of Nuclear Operations and NSG of disagreement between the OSRC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

NOV 1 5 1982

g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15 Rev. 1, February 1979.

j. Modification of Core Protection Calculator (CPC) Addressable Constants.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

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c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics as previously designated by the Station Manager; within 14 days of implementation.

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(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

*See Specification 6.13.1

NOV 15 1982