DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-224 (REV. 4)

This is a request to revise Technical Specification 3/4.7.1.5, "Main Steam Line Isolation Valves," and Technical Specification 3/4.7.1.2, "Auxiliary Feedwater System."

Existing Specifications

Unit 2: See Attachment A
Unit 3: See Attachment B

Proposed Specifications

Unit 2: See Attachment C
Unit 3: See Attachment D

Description

The proposed change would replace in its entirety, Technical Specification 3/4.7.1.5, "Main Steam Line Isolation Valves (MSIV's)," with a new Technical Specification 3/4.7.1.5, "Main Steam and Feedwater Isolation Valves." The proposed change also revises Technical Specification 3/4.7.1.2, "Auxiliary Feedwater System," to explicitly define operability requirements for the auxiliary feedwater isolation and control valves. Technical Specification 3/4.7.1.5 currently defines operability requirements for MSIV's and actions to be taken when one or both MSIV's are inoperable. The operability requirements for the MSIV's ensure that no more than one steam generator will blow down in the event of a main steam line rupture assuming a single failure. Ensuring that only one steam generator blows down prevents the containment design pressure from being exceeded and limits positive reactivity addition due to cooldown of the reactor coolant system. Equally important in mitigating the consequences of these events are the main feedwater isolation valves and other secondary system valves, such as those associated with the auxiliary feedwater system, which are actuated by a main steam isolation signal (MSIS) and/or a containment isolation actuation signal (CIAS).

Technical Specification 3/4.7.1.2 defines operability requirements for the auxiliary feedwater system to ensure that emergency feedwater would be delivered to the steam generators for events requiring the initiation of emergency feedwater for continued secondary heat removal. TS 3/4.7.1.2 currently does not address the function of certain auxiliary feedwater isolation and control valves to close on an MSIS to prevent feeding the affected steam generator during a postulated steam generator rupture, and thereby limiting containment peak pressure and RCS cooldown.

Response times for the above mentioned valves are included as part of overall engineered features actuation system response times in Table 3.3-5, "Engineered Safety Features (ESF) Response Times," of Technical Specification 3/4.3.2, "Engineered Safety Feature Actuation System." However, the TS 3/4.3.2 actions address only instrumentation inoperability and provide no

specific actions when actuated components are inoperable. In most cases other technical specifications limiting conditions for operation (LCO's) address actuated components and provide appropriate action statements. This is currently not the case for the above mentioned valves associated with the secondary system, with the exception of the MSIV's.

The proposed change would add LCO's, surveillance requirements to verify operability, and appropriate actions to be taken which are currently not included in technical specifications for these additional valves.

Specifically, the proposed change would revise Technical Specification 3/4.7.1.5 to include operability requirements for the main feedwater isolation valves (MFIV), main feedwater backup isolation valves (MFBIV) (consisting of one main feedwater regulating valve block valve and one associated regulator valve bypass valve for each steam generator), steam generator sample isolation valve (SGSIV) and blowdown isolation valve (BIV) along with the main steam isolation valves, which are currently addressed by Technical Specification 3/4.7.1.5.

The TS 3/4.7.1.5 action statement would be revised to define specific actions to be taken to address various aspects of inoperability of these valves and recognize inherent redundancy incorporated in the design of the MSIV's and MFIV's.

In the event that an MSIV is inoperable, the action statement currently allows operation with the valve open to continue provided the valve is restored to operable status within four hours. Otherwise, power must be reduced to less than 5% within the next two hours. Thus, operation may continue at above 5% power with an open, inoperable MSIV for a maximum of six hours under the current action statement. The proposed action will require that the valve either be restored to operable status or power be reduced to less than 5% and the valve closed within 6 hours. Thus, under the proposed action operation may continue at above 5% power with an open, inoperable MSIV for a maximum of six hours. The result is the same under both existing and proposed actions.

The existing action fails to differentiate between degrees of inoperability of an MSIV. The MSIV's are maintained open by hydraulic pressure working against pressurized nitrogen gas. The energy stored in the compressed gas provides the motive force for valve closure. The valve is closed by relieving the hydraulic pressure via the hydraulic dump circuits. Either of the two redundant hydraulic dump circuits which are actuated by redundant trains of instrumentation will successfully close the valve. Thus, the MSIV hydraulic dump circuits are totally redundant to one another. In general, where two redundant trains of equipment are provided, technical specification action

statements allow one train to be out of service for a maximum of 72 hours. Consistent with this philosophy a 72 hour action is proposed for the MSIV actuator dump circuits.

The MFIV actuators are of similar design to the MSIV actuators. Like the MSIV's, the redundant dump circuits of the MFIV's are actuated by both trains of both MSIS and CIAS. Accordingly, a 72 hour action statement is proposed for MFIV redundant actuator dump circuits.

The non-safety related main feedwater backup isolation valves are each actuated by one train of a non-safety grade CIAS. The main feedwater backup isolation valves will terminate feedwater flow to the associated steam generator in the event of a postulated secondary system ruptures inside containment. Thus, the main feedwater backup isolation valves provide a degree of redundancy to the MFIV's. Accordingly, a 72 hour action statement is proposed for the MFIV's and main feedwater backup isolation valves.

A 72 hour action statement is also proposed for the steam generator sample isolation and blowdown isolation valves. These valves neither terminate feedwater flow nor do they block a path which would lead to additional containment pressurization. Closure of these valves on an MSIS limits some additional blowdown and associated RCS cooldown in the event of a secondary system rupture. However, their function is of lesser significance in mitigating the consequences of a secondary system rupture than the MSIV's, and a longer out of service time therefore justified.

In the event the prescribed actions cannot be complied with in the specified interval, the proposed change would continue to require that the plant be in at least hot standby within the next six hours and in hot shutdown within the following six hours. The provisions of Specification 3.0.4 which restricts upward mode changes while relying on an action statement, continue to be not applicable for entry into Modes 2 (startup) and 3 (hot standby) with the proposed change.

Surveillance Requirement 4.7.1.5 currently requires that the MSIV's be verified operable in accordance with the In Service Inspection (ISI) program with closure times as specified. The proposed change will expand the applicability of the surveillance to the all of the newly added valves. Response times for the valves are included in Table 3.3-5, "ESFAS Response Times," and will no longer be repeated in TS 3/4.7.1.5. Additionally, the proposed change will require that the closure of the MFIV's and MSIV's will be verified using one of the two independent dump circuits alternately. The proposed surveillance will also require that each valve is actuated to close on its appropriate actuation signal at least once per eighteen months.

Technical Specification 3/4.7.1.2, "Auxiliary Feedwater System," requires that two motor driven and one steam driven auxiliary feedwater pump be operable along with associated flow paths. The proposed change more explicitly defines the flow path requirements and recognizes the dual function of some auxiliary feedwater system valves in either isolating or providing a flow path to a steam generator depending on the situation.

Specifically, the proposed change would incorporate operability requirements for the auxiliary feedwater isolation valves (AFWIV's), auxiliary feedwater control valve (AFWCV), and the associated auxiliary feedwater bypass control valve (AFWBCV). The proposed change would require that all manual valves in the auxiliary feedwater system be in the correct position and all automatic valves be capable of opening or closing upon actuation of EFAS or MSIS, respectively, except the following:

1. the motor-driven auxiliary feedwater pump discharge bypass control valves (HV-4762 and HV-4763), each capable of being closed,

2. the steam turbine-driven auxiliary feedwater pump steam supply isolation valves (HV-8200 and HV-8201) and turbine stop valve (HV-4716), each capable of being opened.

The proposed change would ensure the capability to isolate the auxiliary feedwater system on a MSIS test signal to prevent feeding of the affected steam generator during postulated secondary system rupture events and will continue to ensure the capability to provide emergency feedwater to the appropriate steam generator(s) for secondary heat removal during postulated events where a loss of main feedwater is assumed.

The proposed change revises the TS 3/4.7.1.2 action statements to clarify that the action statement explicitly applies to flow paths as well as auxiliary feedwater pumps.

The proposed change modifies Surveillance Requirement 4.7.1.2.1.b.2 to clarify that only the motor driven AFW pumps start automatically upon receipt of an EFAS test signal. The steam driven turbine pump is routinely tested in accordance with Surveillance Requirement 4.7.1.2.1.a.1 and the inlet valve to this pump is verified to open upon receipt of an EFAS test signal per Surveillance Requirement 4.7.1.2.1.b.1. A new Surveillance Requirement 4.7.1.2.1.b(3) will be added to demonstrate operability by verifying that each automatic valve in the flow path must be in its isolation position on a MSIS test signal except HV-8200 and HV-8201. In addition, operability of all of these valves will be required to be demonstrated at least once per 18 months.

The proposed change will also revise the Bases to Technical Specification 3/4.7.1.5 to cover both the MSIV's and the MFIV's instead of just MSIV's. Specifically, the Bases 3/4.7.1.5 will clarify the functions of the MSIV's and the MFIV's in the event of a main steam or feedwater line rupture. The term "actuation signals" is also added along with the main steam and feedwater isolation system valve closure times specified by the Surveillance Requirements to achieve consistency with the assumptions used in the accident analysis.

Safety Analysis

The proposed changes discussed 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 consequence of an accident previously evaluated?

   Response: No

   The proposed change defines the required LCO's and action statements for the main steam and feedwater isolation valves which currently exist only for the MSIV's. The proposed change also incorporates the required action statements for the auxiliary feedwater isolation and control valves in Technical Specification 3/4.7.1.2. Thus, these changes more clearly define plant operation to be consistent with the assumptions of the accident analyses, thereby avoiding any potentially unacceptable consequences for design basis steam or feedwater line breaks. Therefore, operation of the facility in accordance with this proposed change does not involve a significant increase in 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

   The proposed change does not alter the configuration of the plant or its operation. Therefore, the proposed change does 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 change does not reduce the effectiveness of the main steam and feedwater isolation valves or the auxiliary feedwater system. Therefore, the proposed change will not involve a significant reduction in 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 consideration. Example (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications.
The main steam and feedwater isolation valves, consisting of MSIV's, MFIV's, MFBIV's, SGSIV's, BIV's, and the auxiliary feedwater isolation and control valves, consisting of AFWIV's, AFWCV's and AFWBCV's, are credited in the accident analyses in the mitigation of postulated secondary system ruptures. Currently, only the MSIV's have explicit operability, action, and surveillance requirements defined in technical specifications. Although response times are defined for all these valves by Technical Specification 3/4.3.2, "Engineered Safety Features Actuation System," the TS 3/4.3.2 operability, action, and surveillance requirements are defined only in terms of instrumentation and do not address actuated components.

The proposed change adds new technical specification requirements explicitly addressing operability, action, and surveillance requirements for these valves which do not currently exist within technical specifications. These new requirements constitute additional limitations or restrictions not presently included in technical specifications, therefore, the proposed change is similar to example (ii) of 48 FR 14870.

<u>Safety and Significant Hazards Determination</u>

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; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in margin of safety. In addition, it is considered 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.

Attachment "A"

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 6.0 seconds when tested pursuant to Specification 4.0.5.

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump for entry into MODE 3.

2. 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.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T-121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

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SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that the AFW piping is full of water by venting the accessible discharge piping high points.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an EFAS test signal.

2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

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

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 30°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

Attachment "B"

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 6.0 seconds when tested pursuant to Specification 4.0.5.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

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

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 30°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

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PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine-driven pump for entry into MODE 3.

2. 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.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T-121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

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SURVEILLANCE REQUIREMENTS (Continued)

4. Verifying that the AFW piping is full of water by venting the accessible discharge piping high points.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an EFAS test signal.

2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

Attachment "C"

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses,

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system,

c. Manual valves in the correct position and automatic valves each capable of being opened and closed, except the motor-driven auxiliary feedwater pump discharge bypass control valves (HV-4762 and HV-4763) each capable of being closed; and the steam turbine-driven auxiliary feedwater pump steam supply isolation valves (HV-8200 and HV-8201), and turbine stop valve (HV-4716) each capable of only being opened.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. With any two auxiliary pumps and/or flow paths inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump and flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

2. 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.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

4. Verifying that the AFW piping is full of water by venting the accessible discharge piping high points.

b. At least once per 18 months by:

1. Verifying that each automatic valve actuates to its correct position upon receipt of an EFAS test signal.

2. Verifying that each motor driven pump starts automatically upon receipt of an EFAS test signal.

3. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of a MSIS test signal except HV-8200 and HV-8201.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

MAIN STEAM AND FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 The main steam and feedwater isolation valves shall be OPERABLE for each steam generator with:

a. One OPERABLE main steam isolation valve, with two independent actuator dump circuits.

b. One OPERABLE main feedwater isolation valve, with two independent actuator dump circuits.

c. One OPERABLE set of main feedwater backup isolation valves, consisting of one main feedwater regulating valve block valve and one associated regulating valve bypass valve.

d. One steam generator sample isolation valve, and one blowdown isolation valve, which are either operable or secured in the closed position.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

a. With one main steam isolation valve inoperable, restore the inoperable valve to OPERABLE status, or reduce power to less than or equal to 5 percent of RATED THERMAL POWER and close at least one main steam isolation valve, within 6 hours.

b. With one main feedwater isolation valve or one set of main feedwater backup isolation valves inoperable, restore the inoperable valve(s) to OPERABLE status by securing the inoperable valve(s) in the closed position, or isolate the affected line by use of at least one closed manual valve within 72 hours.

c. With one or more steam generator sample isolation valves or blowdown isolation valves inoperable and open, restore the inoperable valve(s) to OPERABLE status by securing the inoperable valve(s) in the closed position, or isolate the affected line by use of at least one closed manual valve or blind flange within 72 hours.

d. With one main steam isolation valve actuator dump circuit inoperable, restore the inoperable actuator dump circuit to OPERABLE status within 72 hours, or declare the affected main steam isolation valve inoperable.

e. With one main feedwater isolation valve actuator dump circuit inoperable, restore the inoperable actuator dump circuit to OPERABLE status within 72 hours, or declare the affected main feedwater isolation valve inoperable.

f. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

The provisions of Specification 3.0.4 are not applicable for Modes 2 and 3.

SURVEILLANCE REQUIREMENTS

4.7.1.5 The main steam and feedwater isolation valves shall be demonstrated OPERABLE:

a. By verifying closure of main steam and feedwater isolation system valve within the respective limit when tested pursuant to Specification 4.0.5.

   For the main steam isolation valve, closure shall be verified using one of the two independent actuator dump circuits alternately.

b. At least once per 18 months by:

   1. Verifying that each main steam isolation valve, main feedwater isolation valve, steam generator sample isolation valve and blowdown isolation valve actuates to its isolation position on a MSIS test signal.

   2. Verifying that each main steam isolation valve, main feedwater isolation valve and main feedwater backup isolation valve actuates to its isolation position on a CIAS test signal.

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3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM AND FEEDWATER ISOLATION VALVES

The OPERABILITY of the main steam and feedwater isolation valves ensures that the blowdown in the event of a main steam or feedwater line rupture is limited to one steam generator and to an acceptable total mass. This function is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with a main steam blowdown, and (2) limit the pressure rise within containment in the event the main steam or feedwater line rupture occurs within containment. The main steam and feedwater isolation system valve closure times and actuation signals specified by Table 3.3-5 are consistent with the assumptions used in the accident analyses. Although HV-8200, HV-8201, HV-8202 and HV-8203 receive MSIS signals, no operability requirements are imposed upon receipt of this signal because no credit for closure is assumed in the accident analysis.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 30°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

Attachment "D"

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses,

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system,

c. Manual valves in the correct position and automatic valves each capable of being opened and closed, except the motor-driven auxiliary feedwater pump discharge bypass control valves (HV-4762 and HV-4763) each capable of being closed; and the steam turbine-driven auxiliary feedwater pump steam supply isolation valves (HV-8200 and HV-8201), and turbine stop valve (HV-4716) each capable of only being opened.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. With any two auxiliary pumps and/or flow paths inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump and flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump for entry into MODE 3.

SURVEILLANCE REQUIREMENTS (Continued)

2. 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.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

4. Verifying that the AFW piping is full of water by venting the accessible discharge piping high points.

b. At least once per 18 months by:

1. Verifying that each automatic valve actuates to its correct position upon receipt of an EFAS test signal.

2. Verifying that each motor driven pump starts automatically upon receipt of an EFAS test signal.

3. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of a MSIS test signal except HV-8200 and HV-8201.

4.7.1.2.2 The auxiliary feedwater system shall be demonstrated OPERABLE prior to entering MODE 2 following each COLD SHUTDOWN by performing a flow test to verify the normal flow path from the primary AFW supply tank (condensate storage tank T-121) through each auxiliary feedwater pump to its associated steam generator.

MAIN STEAM AND FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 The main steam and feedwater isolation valves shall be OPERABLE for each steam generator with:

a. One OPERABLE main steam isolation valve, with two independent actuator dump circuits.

b. One OPERABLE main feedwater isolation valve, with two independent actuator dump circuits.

c. One OPERABLE set of main feedwater backup isolation valves, consisting of one main feedwater regulating valve block valve and one associated regulating valve bypass valve.

d. One steam generator sample isolation valve, and one blowdown isolation valve, which are either operable or secured in the closed position.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

a. With one main steam isolation valve inoperable, restore the inoperable valve to OPERABLE status, or reduce power to less than or equal to 5 percent of RATED THERMAL POWER and close at least one main steam isolation valve, within 6 hours.

b. With one main feedwater isolation valve or one set of main feedwater backup isolation valves inoperable, restore the inoperable valve(s) to OPERABLE status by securing the inoperable valve(s) in the closed position, or isolate the affected line by use of at least one closed manual valve within 72 hours.

c. With one or more steam generator sample isolation valves or blowdown isolation valves inoperable and open, restore the inoperable valve(s) to OPERABLE status by securing the inoperable valve(s) in the closed position, or isolate the affected line by use of at least one closed manual valve or blind flange within 72 hours.

d. With one main steam isolation valve actuator dump circuit inoperable, restore the inoperable actuator dump circuit to OPERABLE status within 72 hours, or declare the affected main steam isolation valve inoperable.

e. With one main feedwater isolation valve actuator dump circuit inoperable, restore the inoperable actuator dump circuit to OPERABLE status within 72 hours, or declare the affected main feedwater isolation valve inoperable.

f. Otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

The provisions of Specification 3.0.4 are not applicable for Modes 2 and 3.

SURVEILLANCE REQUIREMENTS

4.7.1.5 The main steam and feedwater isolation valves shall be demonstrated OPERABLE:

a. By verifying closure of main steam and feedwater isolation system valve within the respective limit when tested pursuant to Specification 4.0.5.

   For the main steam isolation valve, closure shall be verified using one of the two independent actuator dump circuits alternately.

b. At least once per 18 months by:

   1. Verifying that each main steam isolation valve, main feedwater isolation valve, steam generator sample isolation valve and blowdown isolation valve actuates to its isolation position on a MSIS test signal.

   2. Verifying that each main steam isolation valve, main feedwater isolation valve and main feedwater backup isolation valve actuates to its isolation position on a CIAS test signal.

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3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM AND FEEDWATER ISOLATION VALVES

The OPERABILITY of the main steam and feedwater isolation valves ensures that the blowdown in the event of a main steam or feedwater line rupture is limited to one steam generator and to an acceptable total mass. This function is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with a main steam blowdown, and (2) limit the pressure rise within containment in the event the main steam or feedwater line rupture occurs within containment. The main steam and feedwater isolation system valve closure times and actuation signals specified by Table 3.3-5 are consistent with the assumptions used in the accident analyses. Although HV-8200, HV-8201, HV-8202 and HV-8203 receive MSIS signals, no operability requirements are imposed upon receipt of this signal because no credit for closure is assumed in the accident analysis.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 30°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling system ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGES DPR-13-178 AND NPF-10/15-229

This is a request to revise License Conditions 3.G in DPR-13 and 2.E in NPF-10 and 15.

Existing License Conditions:

Unit 1: See Attachment A
Unit 2: See Attachment B
Unit 3: See Attachment C

Proposed License Conditions:

Unit 1: See Attachment D
Unit 2: See Attachment E
Unit 3: See Attachment F

Description

The proposed changes would revise License Conditions 3.G in DPR-13 and 2.E in NPF-10 and 15. The purpose of these License Conditions is to maintain in effect and fully implement all provisions of the physical security, guard training and qualification and safeguards contingency plans and all amendments and revisions made pursuant to the authority of 10 CFR 50.59 and 10 CFR 50.54(p). (See Attachment G for current list of changes.)

10 CFR 50.54(p) permits licensees to make changes to the Commission-approved security plan which do not involve a reduction in the plan effectiveness without prior NRC approval. Changes to the security plan which may in some way be construed to reduce the effectiveness of the plan or concern changes to vital areas or vital area barriers require prior NRC approval via a license amendment request submitted in accordance with 10 CFR 50.90.

The proposed change would revise license conditions 3.G in DPR-13 and 2.E in NPF-10 and 15 to reflect proposed Revision 14 to the August 1983 Physical Security Plan (PSP). The revised portion of the August 1983 Physical Security Plan concerns a minor change in the location of a vital area boundary which slightly increases the size of a vital area. Details of the proposed revision, provided in Attachment H, are safeguards information and are being withheld from public disclosure pursuant to 10 CFR 73.21.

Safety Analysis

The proposed change discussed 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 an accident previously evaluated?

   Response: No

   The proposed change affects only the Physical Security Plan and is not related to any accident previously evaluated. The proposed change concerns a small change in the location of a vital area boundary to enclose new vital equipment which is required by the NRC to reduce the 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 previously evaluated?

   Response: No

   The proposed change to the Physical Security Plan does not alter any safety related design bases of the facility or its operation. 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 the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change to the Physical Security Plan does not involve a significant reduction in 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. These examples are not applicable to the proposed revision to the August 1983 Physical Security Plan.

Based on the above responses to the three standards of 10 CFR 50.92(c), it can be concluded that the standards are met with a no significant hazards consideration determination.

The proposed change described above, will revise License Condition 3.G in DPR-13 and 2.E in NPF-10 and 15 to reflect the 10 CFR 50.90 submittal of Revision 14 to the August 1983 Physical Security Plan.

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 change; 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.

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ATTACHMENT A

(EXISTING LICENSE CONDITION 3.G)

G. Physical Protection

(1) Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved Physical Security Plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plan which contains 10 CFR 73.21 information is collectively titled "San Onofre Nuclear Generating Station Units 1, 2 and 3 Physical Security Plan," dated August 1983, (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983, (transmitted by letter dated December 16, 1983), Revision 1, dated December 1983 (transmitted by letter dated December 16, 1983) and Revision 1A, dated April 1984 (transmitted by letter dated April 2, 1984). (Originally approved by License Amendment No. 40, dated April 10, 1979). 78 8/7/84

(2) Southern California Edison Company shall fully implement and maintain in effect all provisions of the Commission-approved Safeguards Contingency Plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54. The approved Contingency Plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as Chapter 8 (Revision 1) of the "Physical Security Plan - San Onofre Nuclear Generating Station, Unit 1", as revised August 28, 1980, submitted pursuant to 10 CFR 73.40. The Plan shall be fully implemented in accordance with 10 CFR 73.40(b), within 30 days of this approval by the Commission. 51 1/22/81

(3) Southern California Edison Company shall fully implement and maintain in effect all provisions of the Commission-approved Guard Training and Qualification Plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved plan consists of a document withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as "San Onofre Nuclear Generating Station Guard Training and Qualification Plan", dated September 1980. This plan shall be fully implemented, in accordance with 10 CFR 73.55(b)(4), with 60 days of this approval by the Commission. All security personnel shall be qualified within two years of this approval.

ATTACHMENT B

(EXISTING LICENSE CONDITION 2.E)

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. The Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain 10 CFR 73.21 information are collectively entitled: "San Onofre Nuclear Generating Station Units 1, 2 and 3 Physical Security Plan," dated August 1983, (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983, (transmitted by letter dated December 16, 1983), and Revision 1A, dated April 1984 (transmitted by letter dated April 2, 1984); the Safeguards Contingency Plan (Chapter 8 of the March 1981 Physical Security Plan*); and the "Guard Training and Qualification Plan, San Onofre Nuclear Generating Station Units 1, 2 and 3," dated August 13, 1979 as revised September 3, 1980 and December 15, 1981.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

ATTACHMENT C

(EXISTING LICENSE CONDITION 2.E)

E. The Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved physical security and guard training and qualification plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans which contain 10 CFR 73.21 information are collectively entitled: "San Onofre Nuclear Generating Station Units 1, 2 and 3 Physical Security Plan," dated August 1983, (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983, (transmitted by letter dated December 16, 1983), and Revision 1A, dated April 1984 (transmitted by letter dated April 2, 1984); the Safeguards Contingency Plan (Chapter 8 of the March 1981 Physical Security Plan*); and the "Guard Training and Qualification Plan, San Onofre Nuclear Generating Station Units 1, 2 and 3," dated August 13, 1979 as revised September 3, 1980 and December 15, 1981.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. SCE shall report any violations of the requirements contained in Section 2, Items C.(1), C.(3) through C.(22), E., and F. of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region V, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on October 18, 2013.

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* On September 29, 1983, the Safeguards Contingency Plan was made a separate, companion document to the Physical Security Plan pursuant to the authority of 10 CFR 50.54.

ATTACHMENT D

(PROPOSED LICENSE CONDITION 3.G)

G. Physical Protection

(1) Southern California Edison Company shall maintain in effect and fully implement all provisions of the Commission-approved Physical Security Plan, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plan which contains 10 CFR 73.21 information is collectively titled "San Onofre Nuclear Generating Station Units 1, 2 and 3 Physical Security Plan," dated August 1983, (transmitted by letter dated August 9, 1983), as supported by letter dated October 27, 1983, and as updated by errata page changes dated December 1983, (transmitted by letter dated December 16, 1983), Revision 1, dated December 1983 (transmitted by letter dated December 16, 1983) and Revision 1A, dated April 1984 (transmitted by letter dated April 2, 1984). (Originally approved by License Amendment No. 40, dated April 10, 1979).

(2) Southern California Edison Company shall fully implement and maintain in effect all provisions of the Commission-approved Safeguards Contingency Plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54. The approved Contingency Plan consists of documents withheld from public disclosure pursuant to 10 CFR 2.790(d) identified as Chapter 8 (Revision 1) of the "Physical Security Plan - San Onofre Nuclear Generating Station, Unit 1", as revised August 28, 1980, submitted pursuant to 10 CFR 73.40. The Plan shall be fully implemented in accordance with 10 CFR 73.40(b), within 30 days of this approval by the Commission.

(3) The Southern California Edison Company shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled, "Physical Security Plan," "Safeguards Contingency Plan" and "Security Force Training and Qualification Plan," all subtitled "San Onofre Nuclear Generating Station, Units 1, 2 and 3," and include revisions submitted through July 1987.

ATTACHMENT E

(PROPOSED LICENSE CONDITION 2.E)

D. Exemptions to certain requirements of Appendices G, H and J to 10 CFR Part 50 are described in the Office of Nuclear Reactor Regulation's Safety Evaluation Report. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Therefore, these exemptions are hereby granted. The facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission.

E. The Southern California Edison Company shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled, "Physical Security Plan," "Safeguards Contingency Plan" and "Security Force Training and Qualification Plan," all subtitled "San Onofre Nuclear Generating Station, Units 1, 2 and 3," and include revisions submitted through July 1987.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

ATTACHMENT F

(PROPOSED LICENSE CONDITION 2.E)

E. The Southern California Edison Company shall fully implement and maintain in effect all provisions of the physical security, guard training and qualification, and safeguards contingency plans previously approved by the Commission and all amendments and revisions to such plans made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled, "Physical Security Plan," "Safeguards Contingency Plan" and "Security Force Training and Qualification Plan," all subtitled "San Onofre Nuclear Generating Station, Units 1, 2 and 3," and include revisions submitted through July 1987.

F. This license is subject to the following additional condition for the protection of the environment:

Before engaging in activities that may result in a significant adverse environmental impact that was not evaluated or that is significantly greater than that evaluated in the Final Environmental Statement, SCE shall provide a written notification of such activities to the NRC Office of Nuclear Reactor Regulation and receive written approval from that office before proceeding with such activities.

G. SCE shall report any violations of the requirements contained in Section 2, Items C.(1), C.(3) through C.(22), E., and F. of this license within 24 hours by telephone and confirm by telegram, mailgram, or facsimile transmission to the NRC Regional Administrator, Region V, or his designee, no later than the first working day following the violation, with a written followup report within fourteen (14) days.

H. SCE shall notify the commission, as soon as possible but not later than one hour, of any accident at this facility which could result in an unplanned release of quantities of fission products in excess of allowable limits for normal operation established by the Commission.

I. SCE shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

J. This license is effective as of the date of issuance and shall expire at midnight on October 18, 2013.

ATTACHMENT G

LIST OF CHANGES TO THE FOLLOWING:

1) Physical Security Plan
2) Safeguards Contingency Plan
3) Training and Qualification Plan

PHYSICAL SECURITY PLAN

Revision	Submittal Date	Approval Date
0	8/9/83	8/30/83
1	9/29/83	5/17/85
1A	4/2/84	5/17/85
2	7/25/84	11/27/84
2S	11/12/84	11/27/84
3	1/11/85	6/11/85
3A	5/31/85	6/11/85
4	4/10/85	4/25/85(1)
5	6/20/85	10/29/85
5A	10/23/85	10/29/85
5B	11/15/85	3/24/86
6	12/12/85	3/24/86
7	11/18/85	4/24/86
7A	4/11/86	4/24/86
8	3/3/86	7/3/86
8A	7/16/86	10/23/86
9	5/8/86	9/30/86
9A	9/17/86	9/30/86
10	6/30/86	11/12/86(2)
10A	9/17/86	11/12/86
10B	10/9/86	11/12/86
11	8/22/86	11/13/86
12	10/22/86	3/20/87
12A	3/17/87	3/20/87
13	2/4/87	Under Review
14	4/10/87	Under Review
15	5/29/87	Under Review

SAFEGUARDS CONTINGENCY PLAN

0	12/2/83	8/30/83
1	5/3/85	9/6/85
1A	8/27/85	9/6/85
2	7/11/86	1/9/87(3)
3	10/22/86	3/16/87
4	4/10/87	Under Review

TRAINING AND QUALIFICATION PLAN

0	8/23/79	
1	9/3/80	
2	3/23/83	
3	10/23/84	5/6/85
3A	4/25/85	5/6/85
4	8/27/85	1/14/86
5	10/22/86	3/16/87

(1) Disapproved
(2) One portion to be evaluated as 10 CFR 50.90 change
(3) One portion not approved

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-231

This is a request to revise Technical Specification 3/4.1.3.6. "Regulating CEA Isertion Limits."

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 Figure 3.1-2 of Technical Specification 3/4.1.3.6, "Regulating CEA Insertion Limits." The existing Technical Specification Figure 3.1-2 provides the CEA withdrawal sequence and insertion limits when the Core Operating Limit Supervisory System (COLSS) is in or out of service. The figure also delineates the Short Term and Long Term Steady State Insertion Limits.

The revised Technical Specification figure relaxes the CEA insertion limits at low power levels (at or below 25% power). Relaxation of the Power Dependent CEA Insertion Limits (PDIL) at low power levels will increase flexibility in the determination of Estimated Critical Position (ECP) during plant approach to criticality and will also help reduce the amount of waste water generated during a plant start up. With the current limits, boration may be required before a start up in order to assure that the ECP is within the zero power CEA insertion limits. Less restrictive CEA insertion limits will preclude the need to borate prior to startup thus reducing the volume of waste water.

Safety Analysis

The proposed change discussed 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 an accident previously evaluated?

   Response: No

   The probability or consequences of an accident is not increased by the proposed change since the plant design is not changed and the revised PDIL does not significantly affect the San Onofre safety analysis consequences.

The following safety analysis events were evaluated with respect to the change:

1. Steam line break (SLB) at zero power
2. CEA withdrawal (CEAW) at low power
3. CEA Ejection (CEAE) at zero power

A critical parameter in the steam line break analysis at zero power is the initial shutdown margin assumed for the event. The zero power SLB reference analysis (Cycle 1) assumed a shutdown margin of 5.15% delta rho (the Technical Specification minimum) for this condition. The revised PDIL results in a lower calculated shutdown margin at the hot zero power critical condition than that for the old PDIL but still greater than 5.15% delta rho. Thus the reference analysis remains bounding.

The revised PDIL results in a higher maximum reactivity insertion rate (1.7×10^{-4} vs. 1.1×10^{-4} delta rho/sec) during a CEA withdrawal event at low power than previously reported. Previous analysis of this event (Cycle 3 RAR) found that an intermediate insertion rate resulted in the most adverse results. A check case determined that this remains true. Thus, the previous analysis results were demonstrated to remain bounding.

The CEA Ejection event at 0% power is most sensitive to ejected rod worth. The ejected rod worth was higher for the new PDIL than for Cycle 3 but less than the value assumed in the reference analysis (Cycle 1). This is due largely to the fact that the Cycle 1 PDIL allowed greater rod insertion at zero power than does either the old or the new PDIL. Thus, the reference analysis was found to remain bounding.

In conclusion, the probability of an accident is not affected since the plant design is not changed. The consequences of a previously analyzed accident are not increased as shown above since the existing safety analyses are demonstrated to remain bounding.

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

There is no change to the probability or consequences of a new or different kind of accident in that the facility design remains unchanged. The proposed change has the effect of allowing more rods in the core at low power relative to Cycle 3 but less than that allowed for Cycle 1. The asymmetric CEA related event consequences are not significantly affected as discussed above.

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

   Response: No

   There is no reduction in the margin of safety previously established. The analysis discussed above has demonstrated that the new PDIL has an insignificant effect on safety analysis results and that previous results are still bounding.

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 (vi) relates to a change which either may result in some increase in 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 refinement of a previously used calculational model or design method. The acceptance criteria of SRP Section 15.1.5 require that short-term and long-term coolability be achieved by confirming that the RCS is maintained in a safe status. The criteria also specifies that, with respect to steam line breaks, the RCS be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded and capability to cool the core is maintained. The acceptance criteria of SRP Section 15.4.1 relative to CEA withdrawal events, requires that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences. The acceptance criteria also require that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system. The specified acceptable fuel design limits are assumed to be met when thermal margin limits (DNBR) are met and fuel centerline temperatures do not exceed the melting point. Relative to CEA ejection, SRP Section 15.4.8 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor cause sufficient damage to impair significantly the capacity to cool the core. These criteria were addressed in the existing safety analysis and since that analysis remains bounding for the proposed change, the SRP acceptance criteria are deemed to be adequately addressed. The proposed change is, therefore, similar to Example (vi) in that the Technical Specification relaxes the CEA insertion limits at low power levels and may result in an insignificant increase in the consequences of a previously analyzed accident, but where the results of the change are clearly within all acceptable criteria with respect to the Standard Review Plan.

Safety and Significant Hazards Determination

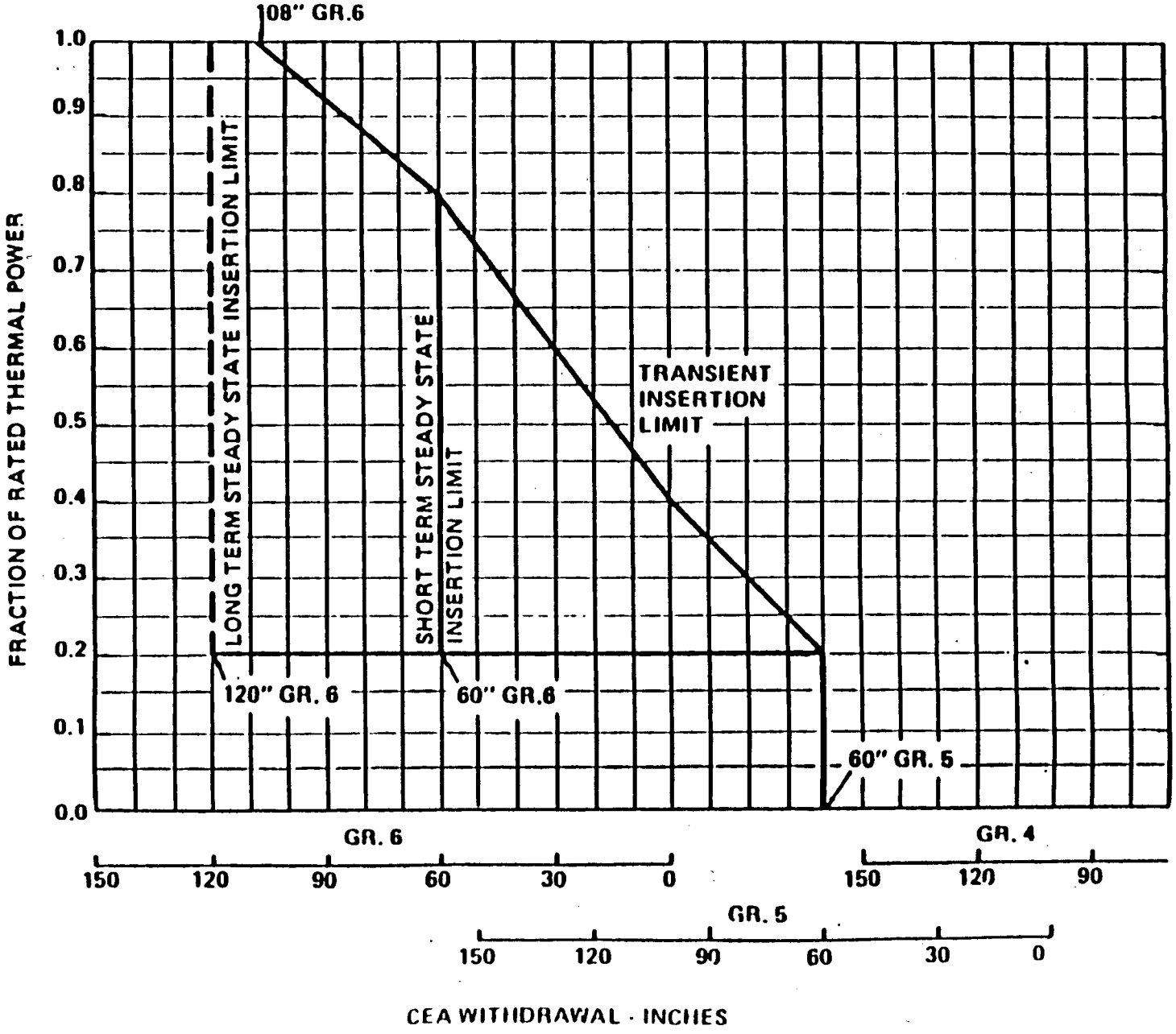
Based on the above Safety Analysis, it is concluded that; (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed cange; and (3) this action 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.

BRD:8463F

ATTACHMENT A

EXISTING TECHNICAL SPECIFICATIONS, UNIT 2

FIGURE 3.1-2



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ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATIONS, UNIT 2

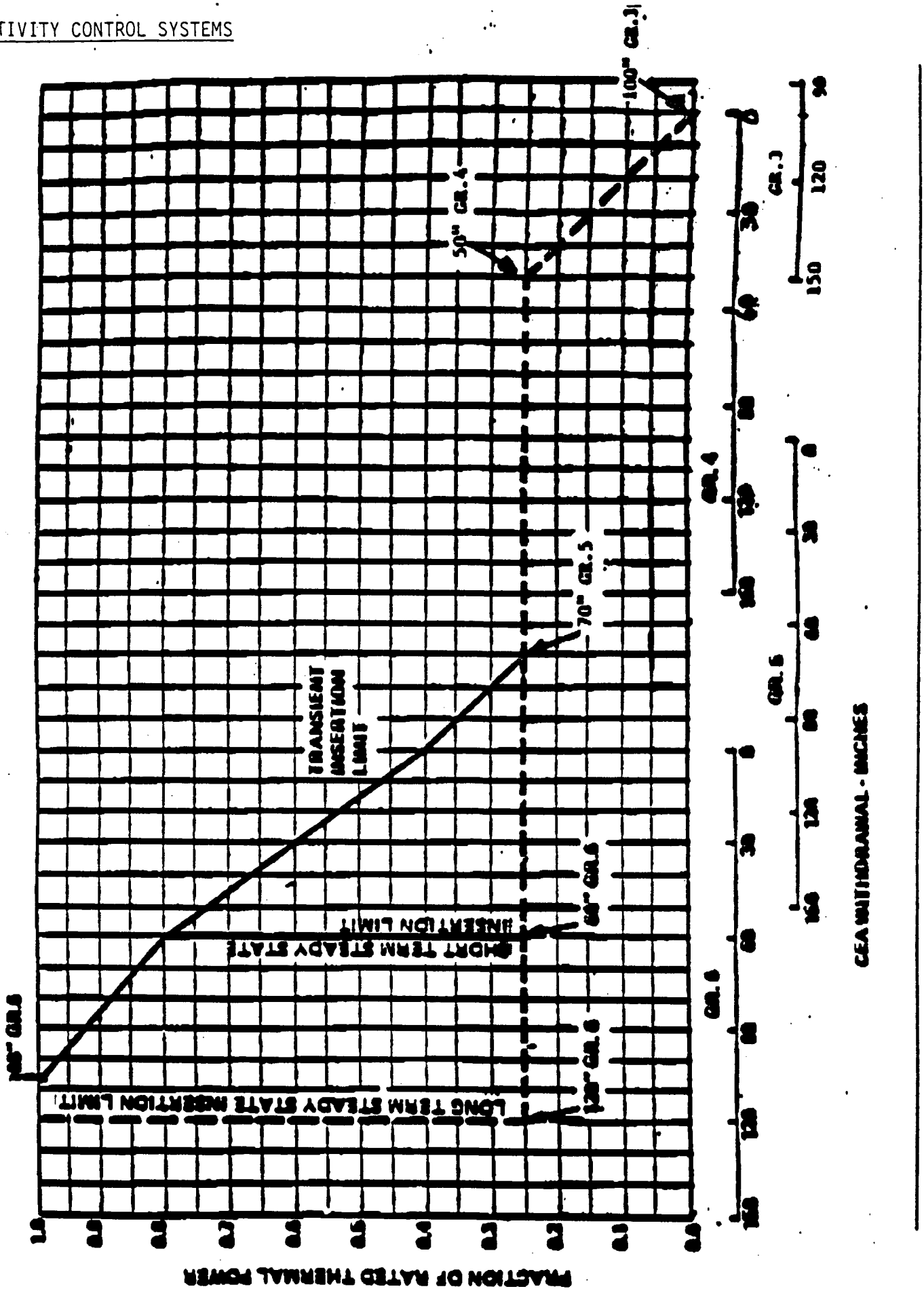
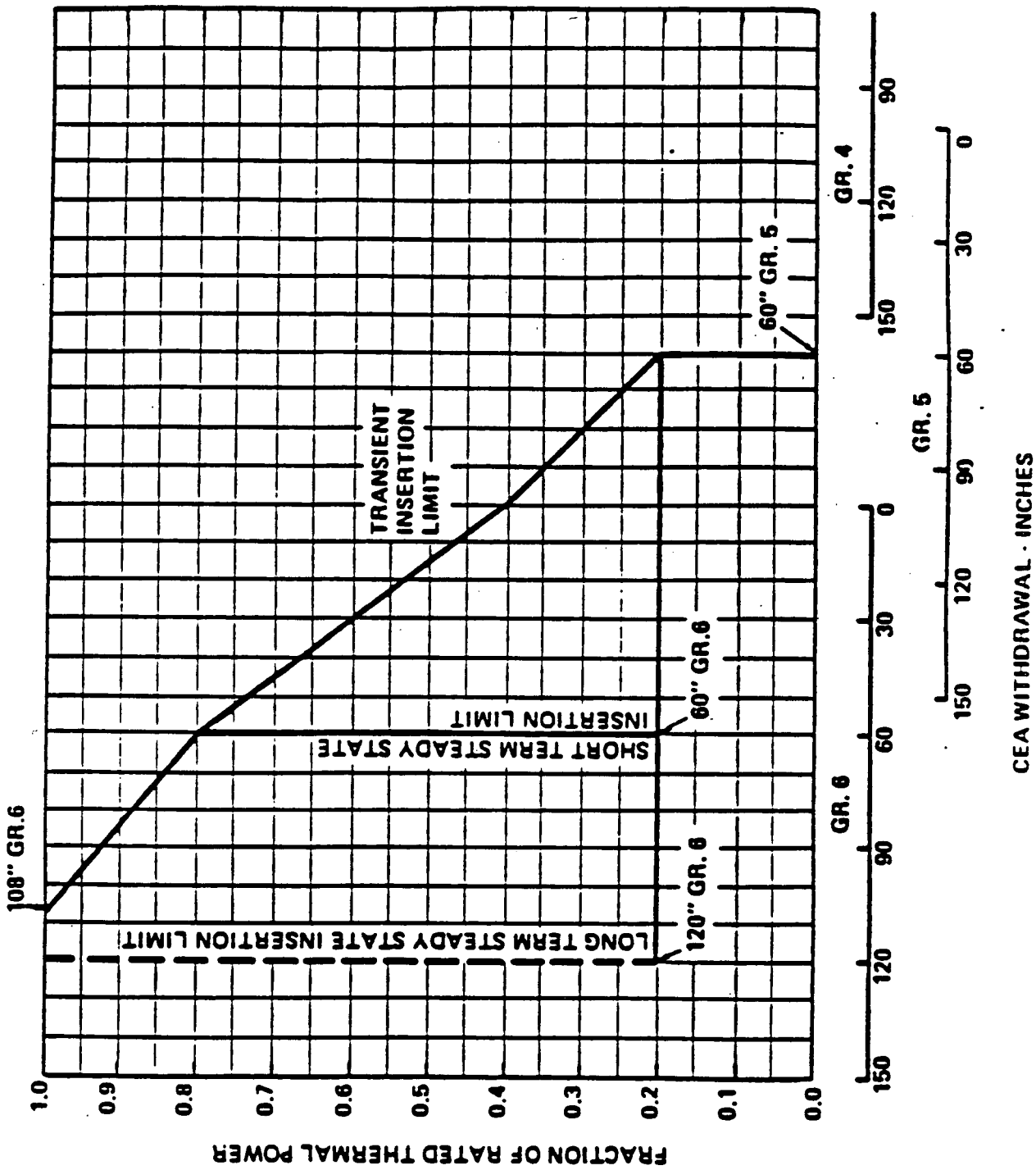

100" GR.3
50" GR.4
70" GR.5
TRANSIENT INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
LONG TERM STEADY STATE INSERTION LIMIT
GR.3
GR.4
GR.5
GR.6
CEA WITHDRAWAL - INCHES
FRACTION OF RATED THERMAL POWER

FIGURE 3.1-2

ATTACHMENT C

EXISTING TECHNICAL SPECIFICATIONS, UNIT 3

FIGURE 3.1-2



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ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATIONS, UNIT 3

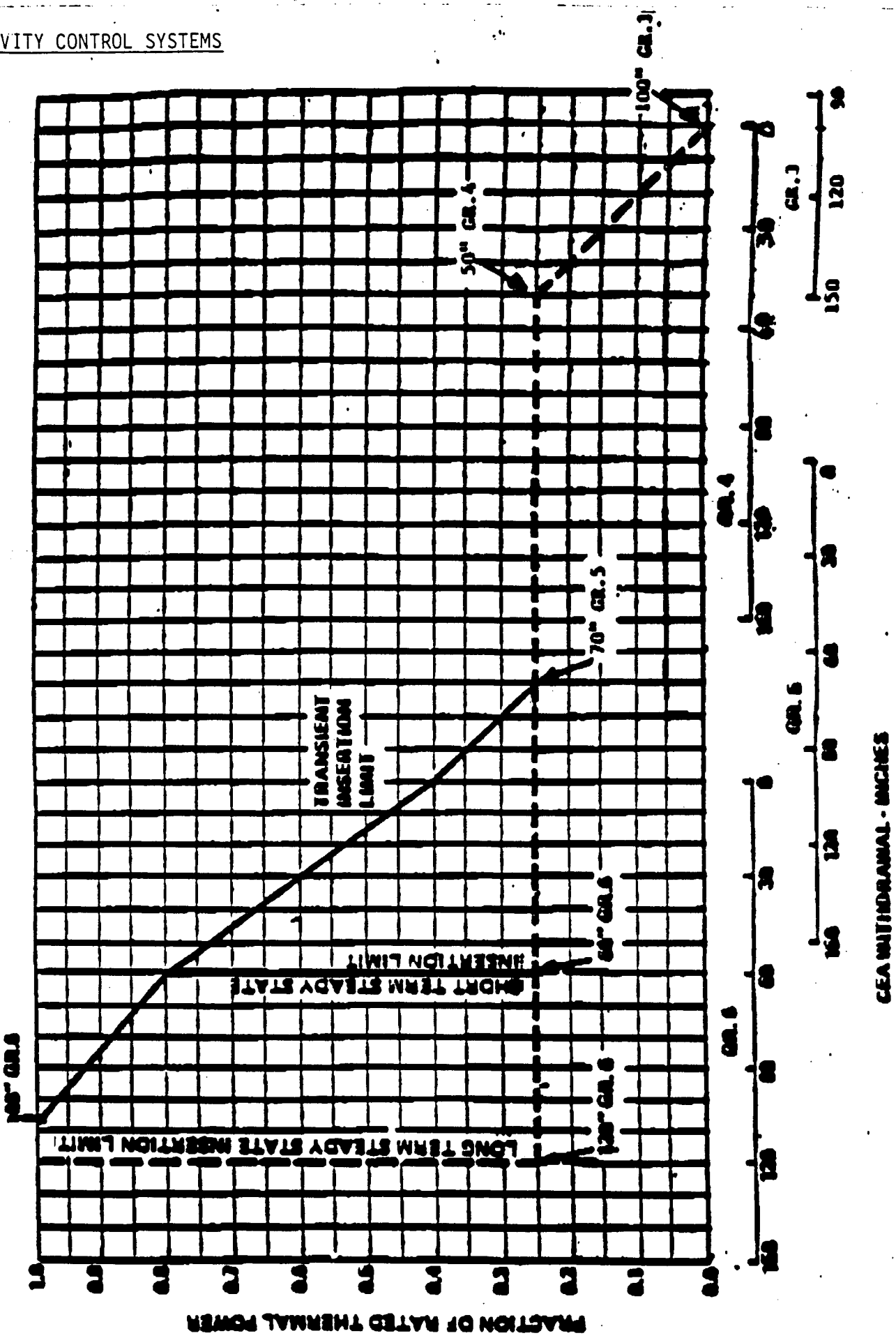

LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
TRANSIENT INSERTION LIMIT
120" GRP. 6
64" GRP. 6
70" GRP. 5
50" GRP. 4
100" GRP. 3
GRP. 6
GRP. 5
GRP. 4
GRP. 3
CEA WITHDRAWAL - INCHES
FRACTION OF RATED THERMAL POWER

FIGURE 3.1-2

PROPOSED LICENSE AMENDMENT

SAN ONOFRE NUCLEAR GENERATING STATION - UNIT NO. 2 and 3

PROPOSED CHANGE NO. NPF-10/15-231

1. DESCRIPTION: The proposed change revises Figure 3.1-2 of Technical Specification 3/4.1.3.6, "Regulatory CEA Insertion Limits." The proposed change relaxes the CEA Insertion Limits at low power levels to increase operating flexibility and reduce the volume of radioactive waste water.

ATTACHED INFORMATION PREPARED BY: B. R. Duncil 21787 6/08/87
Name (PAX) Date

REVIEWED BY: D. L. Cox 21658 6/08/87
Licensing Supervisor (PAX) Date

2. STATION INTERDISCIPLINARY REVIEW

Required		Signature	Date
X	Operations		
	Maintenance		
	Health Physics		
X	Technical		
	Chemistry		
	NSSS		
	Instrumentation and Control		
X	STA		
	NSSS Support		
	Emergency Preparedness		
X	Other (Computer)		

SCE 26-185 REV 12/85

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-231

This is a request to revise Technical Specification 3/4.1.3.6. "Regulating CEA Isertion Limits."

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 Figure 3.1-2 of Technical Specification 3/4.1.3.6, "Regulating CEA Insertion Limits." The existing Technical Specification Figure 3.1-2 provides the CEA withdrawal sequence and insertion limits when the Core Operating Limit Supervisory System (COLSS) is in or out of service. The figure also delineates the Short Term and Long Term Steady State Insertion Limits.

The revised Technical Specification figure relaxes the CEA insertion limits at low power levels (at or below 25% power). Relaxation of the Power Dependent CEA Insertion Limits (PDIL) at low power levels will increase flexibility in the determination of Estimated Critical Position (ECP) during plant approach to criticality and will also help reduce the amount of waste water generated during a plant start up. With the current limits, boration may be required before a start up in order to assure that the ECP is within the zero power CEA insertion limits. Less restrictive CEA insertion limits will preclude the need to borate prior to startup thus reducing the volume of waste water.

Safety Analysis

The proposed change discussed 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 an accident previously evaluated?

   Response: No

   The probability or consequences of an accident is not increased by the proposed change since the plant design is not changed and the revised PDIL does not significantly affect the San Onofre safety analysis consequences.

The following safety analysis events were evaluated with respect to the change:

1. Steam line break (SLB) at zero power
2. CEA withdrawal (CEAW) at low power
3. CEA Ejection (CEAE) at zero power

A critical parameter in the steam line break analysis at zero power is the initial shutdown margin assumed for the event. The zero power SLB reference analysis (Cycle 1) assumed a shutdown margin of 5.15% delta rho (the Technical Specification minimum) for this condition. The revised PDIL results in a lower calculated shutdown margin at the hot zero power critical condition than that for the old PDIL but still greater than 5.15% delta rho. Thus the reference analysis remains bounding.

The revised PDIL results in a higher maximum reactivity insertion rate (1.7×10^{-4} vs. 1.1×10^{-4} delta rho/sec) during a CEA withdrawal event at low power than previously reported. Previous analysis of this event (Cycle 3 RAR) found that an intermediate insertion rate resulted in the most adverse results. A check case determined that this remains true. Thus, the previous analysis results were demonstrated to remain bounding.

The CEA Ejection event at 0% power is most sensitive to ejected rod worth. The ejected rod worth was higher for the new PDIL than for Cycle 3 but less than the value assumed in the reference analysis (Cycle 1). This is due largely to the fact that the Cycle 1 PDIL allowed greater rod insertion at zero power than does either the old or the new PDIL. Thus, the reference analysis was found to remain bounding.

In conclusion, the probability of an accident is not affected since the plant design is not changed. The consequences of a previously analyzed accident are not increased as shown above since the existing safety analyses are demonstrated to remain bounding.

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

There is no change to the probability or consequences of a new or different kind of accident in that the facility design remains unchanged. The proposed change has the effect of allowing more rods in the core at low power relative to Cycle 3 but less than that allowed for Cycle 1. The asymmetric CEA related event consequences are not significantly affected as discussed above.

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

   Response: No

   There is no reduction in the margin of safety previously established. The analysis discussed above has demonstrated that the new PDIL has an insignificant effect on safety analysis results and that previous results are still bounding.

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 (vi) relates to a change which either may result in some increase in 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 refinement of a previously used calculational model or design method. The acceptance criteria of SRP Section 15.1.5 require that short-term and long-term coolability be achieved by confirming that the RCS is maintained in a safe status. The criteria also specifies that, with respect to steam line breaks, the RCS be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded and capability to cool the core is maintained. The acceptance criteria of SRP Section 15.4.1 relative to CEA withdrawal events, requires that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences. The acceptance criteria also require that the reactor protection system be designed to assure that specified acceptable fuel design limits are not exceeded in the event of a single malfunction of the reactivity control system. The specified acceptable fuel design limits are assumed to be met when thermal margin limits (DNBR) are met and fuel centerline temperatures do not exceed the melting point. Relative to CEA ejection, SRP Section 15.4.8 requires that the effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor cause sufficient damage to impair significantly the capacity to cool the core. These criteria were addressed in the existing safety analysis and since that analysis remains bounding for the proposed change, the SRP acceptance criteria are deemed to be adequately addressed. The proposed change is, therefore, similar to Example (vi) in that the Technical Specification relaxes the CEA insertion limits at low power levels and may result in an insignificant increase in the consequences of a previously analyzed accident, but where the results of the change are clearly within all acceptable criteria with respect to the Standard Review Plan.

Safety and Significant Hazards Determination

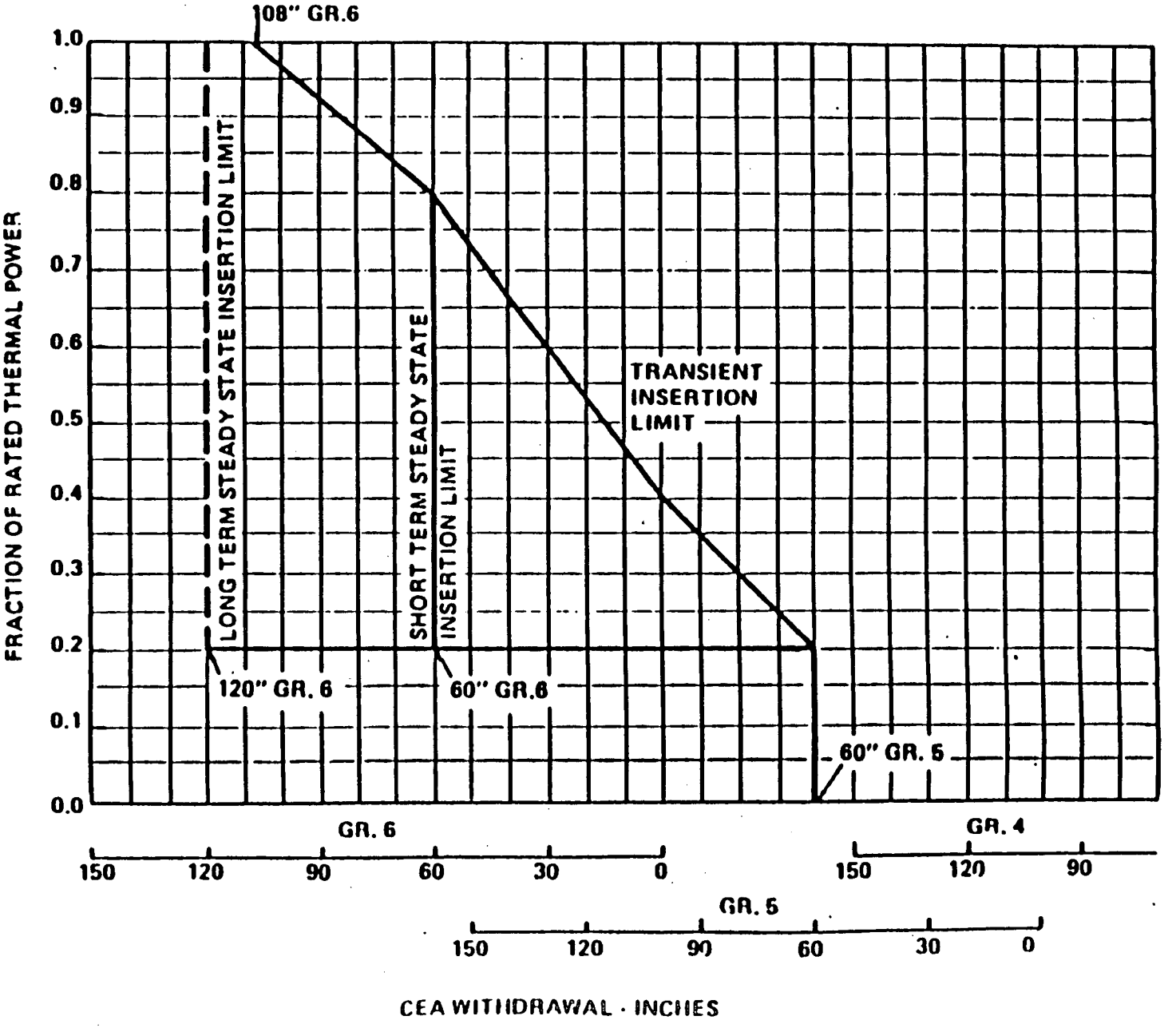
Based on the above Safety Analysis, it is concluded that: (1) the proposed change does not constitute a significant hazards consideration as defined by 10 CFR 50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed cange; and (3) this action 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.

BRD:8463F

ATTACHMENT A

EXISTING TECHNICAL SPECIFICATIONS, UNIT 2

FIGURE 3.1-2


108" GR.6
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
FRACTION OF RATED THERMAL POWER
LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
TRANSIENT INSERTION LIMIT
120" GR. 6
60" GR.6
60" GR. 5
GR. 6
150 120 90 60 30 0
GR. 4
150 120 90
GR. 5
150 120 90 60 30 0
CEA WITHDRAWAL - INCHES

ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATIONS, UNIT 2

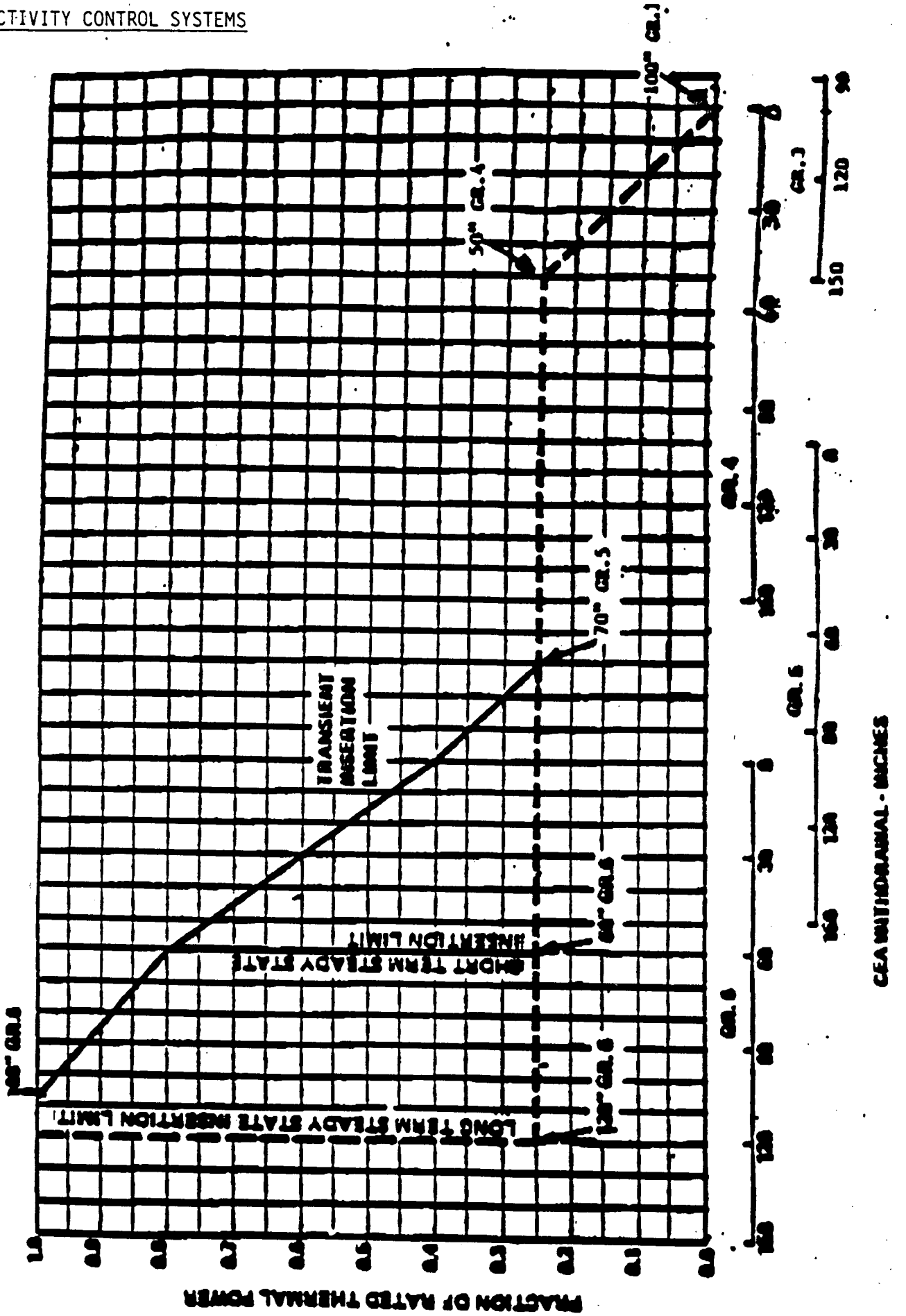
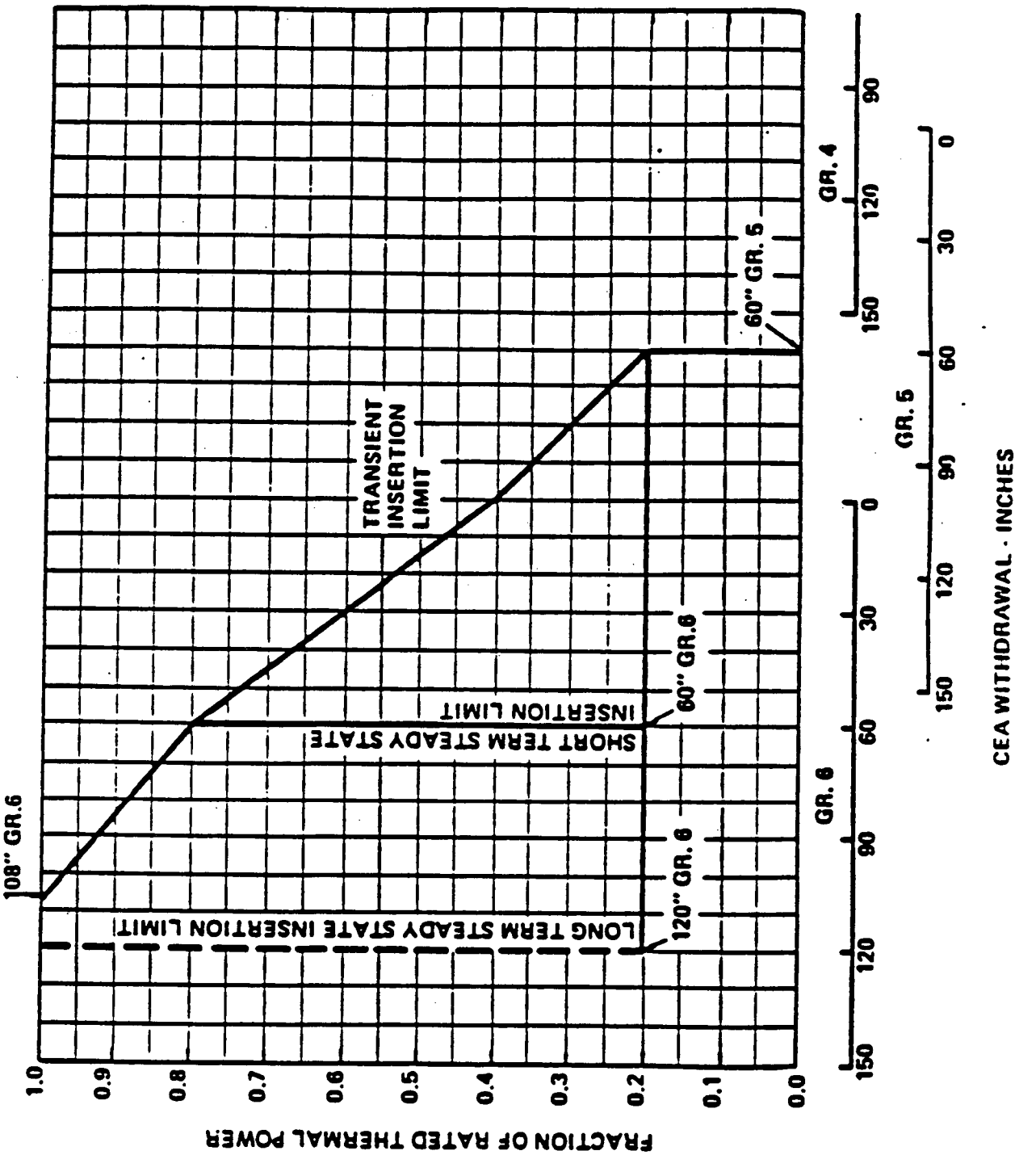

FRACTION OF RATED THERMAL POWER
LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
TRANSIENT INSERTION LIMIT
CEA WITHDRAWAL - INCHES
CRL.1
CRL.2
CRL.3
CRL.4
CRL.5
CRL.6
CRL.7
CRL.8
100" CRL.1
50" CRL.4
70" CRL.5
60" CRL.6
30" CRL.6
150
120
90
60
30
0

FIGURE 3.1-2

ATTACHMENT C

EXISTING TECHNICAL SPECIFICATIONS, UNIT 3

FIGURE 3.1-2



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ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATIONS, UNIT 3

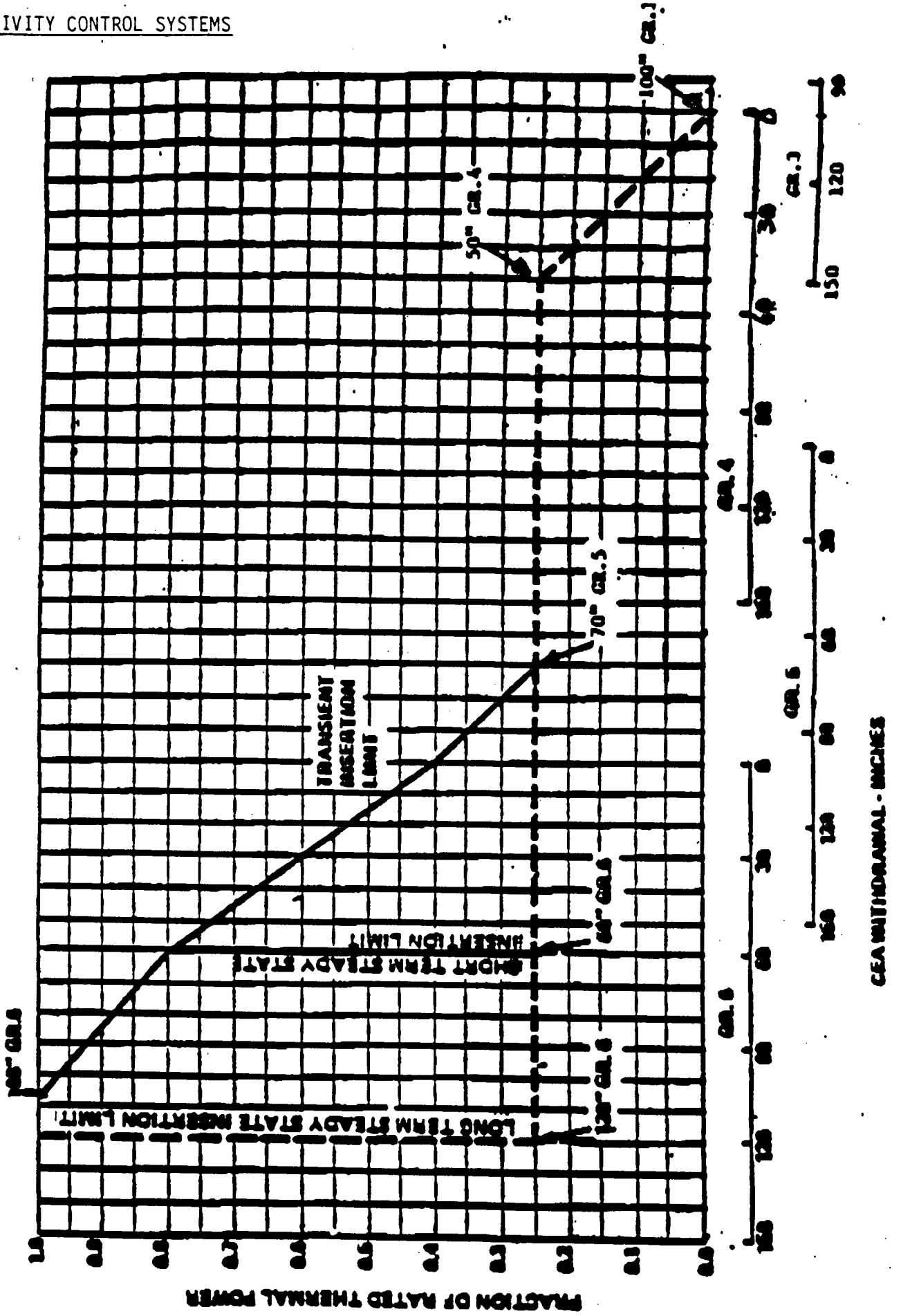
FRACTION OF RATED THERMAL POWER
1.0
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0.0
LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
TRANSIENT INSERTION LIMIT
100" CR.1
50" CR.4
70" CR.5
CR.1
CR.4
CR.5
CEA WITHDRAWAL - INCHES

FIGURE 3.1-2

June 17, 1987

Reviewers of Proposed Change NPF-10/15-231

Please review the attached proposed changes to the Units 2/3 Technical Specifications by June 24, 1987. It is anticipated that implementation responsibility will be as follows:

T. S. SECTION	SPECIFICATION	CHANGE	ORGANIZATION
Figure 3.1-2	Regulating CEA Insertion Limits	Insertion limits	Operations

Contact me if you have any questions concerning the attached document, or call Char Prince (PAX 89805) for pickup after review.

Jaime Elliot

JAIME ELLIOT

JE:9976u:kf
Attachment

cc: R. J. Maisel (w/o attachment)
P. L. Jones
T. J. Vogt
D. H. Peacor

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-231

This is a request to revise Technical Specification 3/4.1.3.6. "Regulating CEA Isertion Limits."

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 Figure 3.1-2 of Technical Specification 3/4.1.3.6, "Regulating CEA Insertion Limits." The existing Technical Specification Figure 3.1-2 provides the CEA withdrawal sequence and insertion limits when the Core Operating Limit Supervisory System (COLSS) is in or out of service. The figure also delineates the Short Term and Long Term Steady State Insertion Limits.

The revised Technical Specification figure relaxes the CEA insertion limits at low power levels (at or below 25% power). Relaxation of the Power Dependent CEA Insertion Limits (PDIL) at low power levels will increase flexibility in the determination of Estimated Critical Position (ECP) during plant approach to criticality and will also help reduce the amount of waste water generated during a plant start up. With the current limits, boration may be required before a start up in order to assure that the ECP is within the zero power CEA insertion limits. Less restrictive CEA insertion limits will preclude the need to borate prior to startup thus reducing the volume of waste water.

Safety Analysis

The proposed change discussed 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 an accident previously evaluated?

   Response: No

   The probability or consequences of an accident is not increased by the proposed change since the plant design is not changed and the revised PDIL does not significantly affect the San Onofre safety analysis consequences.

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Response: No

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

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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; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed cange; 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.

BRD:8463F

ATTACHMENT B

PROPOSED TECHNICAL SPECIFICATIONS, UNIT 2

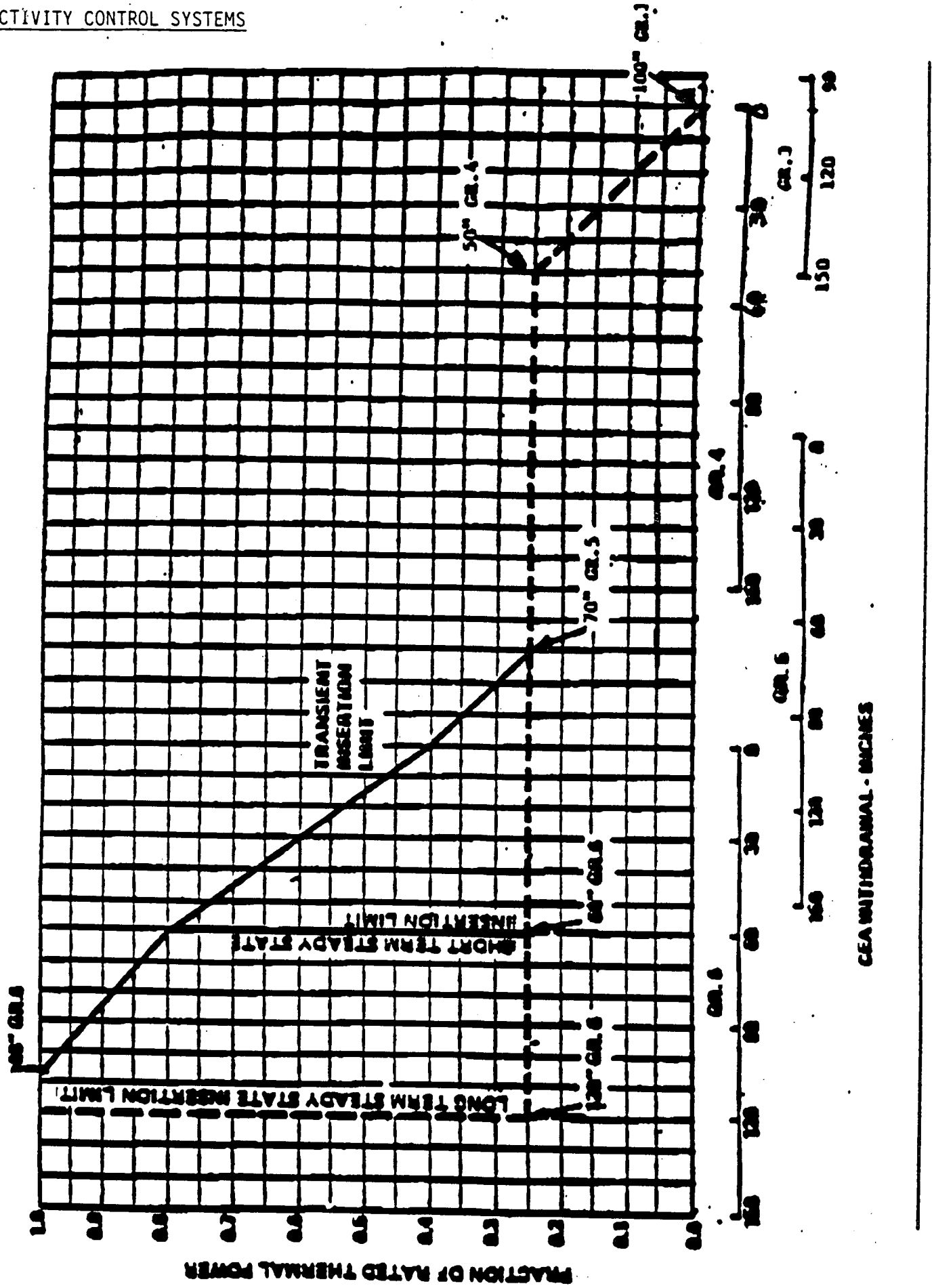
FRACTION OF RATED THERMAL POWER
CEA WITHDRAWAL - INCHES
LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
TRANSIENT INSERTION LIMIT
CRL.1
CRL.2
CRL.3
CRL.4
CRL.5
CRL.6
100" CRL.1
50" CRL.4
70" CRL.5
60" CRL.6
120" CRL.6

FIGURE 3.1-2

ATTACHMENT D

PROPOSED TECHNICAL SPECIFICATIONS, UNIT 3

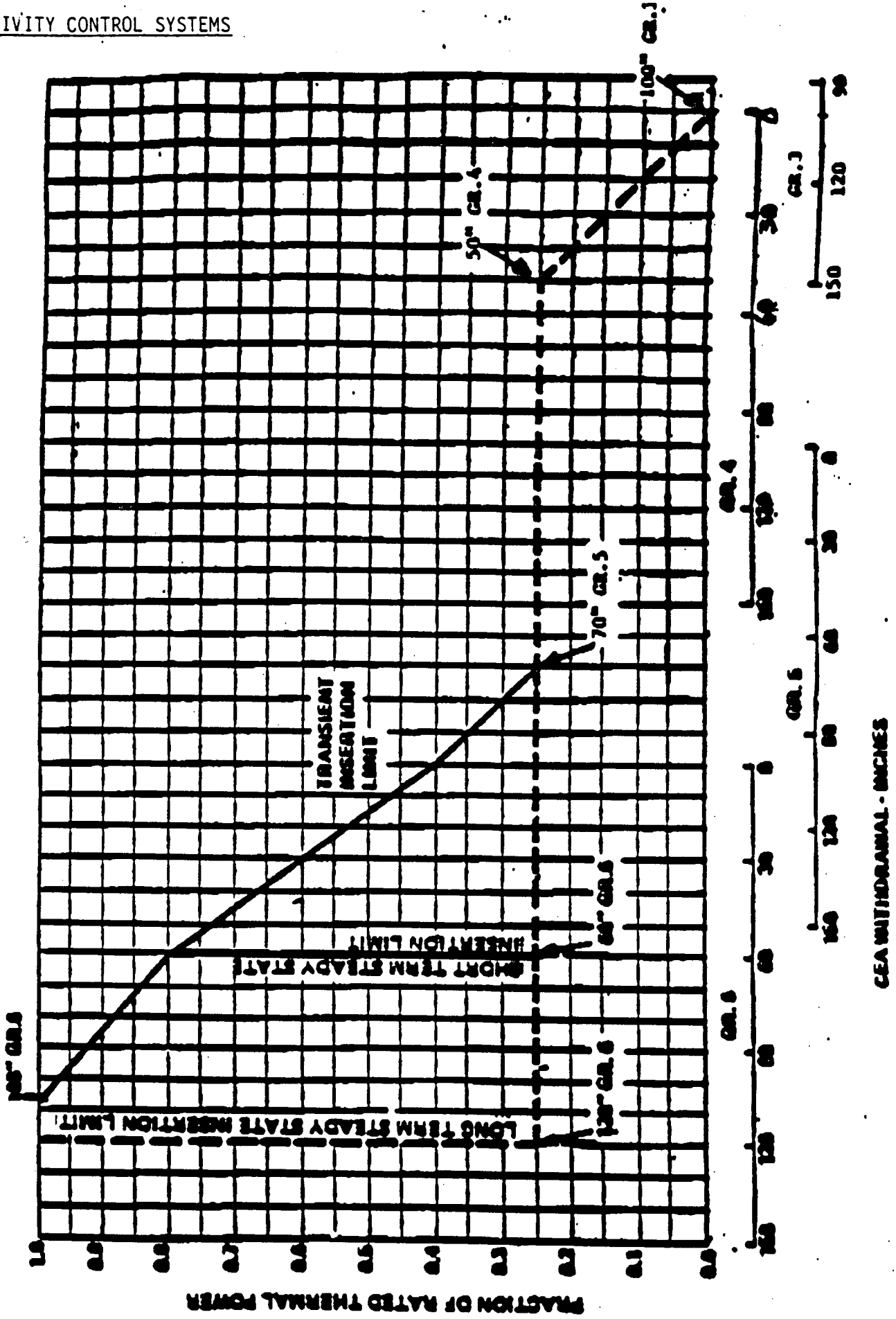
FRACTION OF RATED THERMAL POWER
CEA WITHDRAWAL - INCHES
LONG TERM STEADY STATE INSERTION LIMIT
SHORT TERM STEADY STATE INSERTION LIMIT
TRANSIENT INSERTION LIMIT
20" GR.6
30" GR.6
40" GR.6
50" GR.4
70" GR.5
100" GR.3
GR.6
GR.5
GR.4
GR.3

FIGURE 3.1-2

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-232

This is a request to revise Technical Specification 3/4.3.3, "Monitoring Instrumentation".

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 would revise Surveillance Requirement 4.3.3.2 of Technical Specification 3.3.3.2, "Incore Detectors." The existing Technical Specification defines the operability requirements for the incore detection system. The surveillance requirements identify the system tests and the frequency with which they are to be performed. The performance of these surveillance tests demonstrate the operability of the detection system. The purpose of the Specification is to ensure that the measurements obtained from the use of this system accurately represent the nuclear conditions within the reactor core.

Surveillance Requirement 4.3.3.2(a) requires that the incore detection system be demonstrated operable by performance of a channel check within 24 hours prior to its use if 7 or more days have elapsed since the previous check and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin. Thus, the channel check is required weekly independent of the parallel surveillances required for monitoring the other parameters listed above. Technical Specification 3/4.2.2, "Planar Radial Peaking Factors - Fxy," requires that the measured PLANAR RADIAL PEAKING FACTORS ($F^m xy$) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS ($F^c xy$) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC) when the reactor is in MODE 1 (critical) above 20% of RATED THERMAL POWER. The measured PLANAR RADIAL PEAKING FACTORS ($F^m xy$) are obtained by using the incore detection system after each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and at least once every 31 Effective Full Power Days (EFPD). Technical Specification 3/4.2.3, "Azimuthal Power Tilt - Tq," requires that the AZIMUTHAL POWER TILT (Tq) be less than or equal to the AZIMUTHAL POWER TILT allowance used in the CPC when the reactor is in MODE 1 above 20% of RATED THERMAL POWER. Similarly, Surveillance Requirement 4.2.3(c) requires that the AZIMUTHAL POWER TILT be determined by using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT. The proposed change to Surveillance Requirement 4.3.3.2(a) would change

the frequency of performance of the channel check to within 24 hours of its use if 31 EFPD or more had elapsed since the previous check and at least once per 31 EFPD thereafter when required for monitoring the above listed parameters. The proposed change allows verification of incore detector operability to be performed in conjunction with other routine surveillances thereby greatly decreasing the surveillance workload.

Safety Analysis

The proposed change described 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 incore detectors are utilized by COLSS to verify that Core Axial Shape Index, Linear Heat Rate, and Azimuthal Tilt are within the ranges assumed as beginning points for Anticipated Operational Occurrences and Accidents. The detectors provide no direct protective action. Incore detector reliability studies at SONGS and at least two other utilities indicate that the mean time between incore detector failures during operation is in excess of 100 EFPD. COLSS has internal range checks which reject a detector signal of less than about 5% power or in excess of about 200% power. If the number of failed detectors exceed 25%, COLSS annunciates an alarm to ensure compliance with Technical Specification 3.3.3.2a.

   Extending the surveillance interval from 7 days to 31 EFPD could marginally increase the probability of operating with a detector which would fail its channel check. If a detector fails its channel check, it is manually removed from scan. Normally, COLSS will perform exactly the same function by removing the detector signal from its calculations on a failed range check.

   CE analog plants use fixed incore detectors to directly monitor core linear heat rate and consequently, individual detector signals are of importance. CE digital plants, such as San Onofre Units 2 and 3, monitor DNBR and Linear Heat rate based on detector signals from throughout the core. Thus, the failure of any single detector is of no significant consequence and is analyzed for in setting the 75% lower limit on the number of operable detectors.

   Occasionally a detector will suffer a "soft" failure such that it is no longer generating an accurate signal and yet will not present a large enough deviation for COLSS to reject the signal. This has a limited effect on the COLSS calculations which might be non-conservative.

The CPCs assume an azimuthal tilt allowance in their calculations. COLSS annunciates an alarm if the incore detectors indicate a tilt in excess of the CPC tilt allowance. Typically, the tilt allowance is 2 to 3% higher than actual core tilt. If a soft incore detector failure occurs which causes indicated tilt to increase, COLSS will annunciate an alarm well before actual tilt would require it. If a soft incore detector failure occurs which causes indicated tilt to decrease, the CPCs continue to utilize the original conservative tilt allowance. The amount of tilt changes as a result of a soft failure would be less than 1% power at full power in any case. By Station procedures, Operations personnel are allowed to raise the CPC tilt allowance but prior to decreasing the allowance, Engineering personnel must be consulted to ensure the tilt allowance installed in the CPCs remains conservative relative to the actual core tilt.

Incore detectors are also used to calculate Axial Shape Index. This parameter is averaged over all the operable strings so that the effect of a soft failure in one string is further diluted by the remaining operable strings. The maximum deviation in ASI as a result of a soft detector failure is about 0.016 at 20% power and 0.005 at 100% power. This corresponds to a change in DNBR margin of about 0.8% overpower margin at 100% and 2.4% overpower margin at 20%. Actual studies on SONGS 2 and 3 where single artificial incore signals were injected into the COLSS calculations showed that soft failures at 100% power affect calculated DNBR overpower margin by no more than about 0.5% power. Calculated LPD margin changed by no more than 1% power, but CPC plants are almost invariably DNBR limited.

San Onofre Units 2 and 3 operators continuously monitor ASI and ensure that the axial power oscillations stay small. Should an incore detector fail and cause a significant change in ASI, it would immediately be noticed by the operators. Station Engineering also monitors ASI and would detect any significant step change. Should the ASI change not be immediately noticed, the operators would attempt to maintain an erroneous equilibrium shape index. This inability to maintain a stable axial shape with all rods out would indicate the presence of a problem with corrective action to ensue.

These factors ensure that the unlikely event of a soft detector failure will not change the probability or consequences of a previously analyzed accident.

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; therefore, the proposed change does 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 the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The mean time between incore detector failures is greater than 100 EFPD which is well beyond the proposed 31 EFPD surveillance frequency. The unit is analyzed for up to 25% of the incore detector strings being inoperable. In addition, COLSS automatically provides a conservative check of this requirement. Thus, the proposed change does not involve a significant reduction in 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 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 refinement of a previously used calculation model or design method. Standard Review Plan Section 7.2, Reactor Trip System, requires that the system be designed to initiate automatically the reactivity control system (control rods) to assure that specified acceptable fuel design limits are not exceeded. Increasing the incore detection system channel check frequency from 7 day to 31 EFPD is similar to Example (vi) in that the proposed change may result in an insignificant reduction in the margin of safety, but where the results of the change are clearly within all acceptance criteria as described above with respect to the reactor trip system specified in SRP Section 7.2.

<u>Safety and Significant Hazards Determination</u>

Based on the above Safety Analysis, it is concluded that (1) the proposed change does not constitute a significant hazards consideration as defined by 10CFR50.92; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; 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.

BRD:8517F

ATTACHMENT "A"

Existing Unit-2 Technical Specification

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

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

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if 7 or more days have elasped since the previous check and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

ATTACHMENT "B"

Proposed Unit-2 Technical Specification

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

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

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if 31 EFPD or more have elapsed since the previous check and at least once per 31 EFPD thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

ATTACHMENT "C"

Existing Unit-3 Technical Specification

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

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

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if 7 or more days have elapsed since the previous check and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

ATTACHMENT "D"

Proposed Unit-3 Technical Specification

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

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

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use if 31 EFPD or more have elapsed since the previous check and at least once per 31 EFPD thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-233

This is a request to revise Technical Specification 3/4.10.2, "Group Height, Insertion and Power Distribution Limits."

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 would revise Technical Specification 3/4.10.2, "Group Height, Insertion and Power Distribution Limits," and Tables 2.2-1 and 3.3-1. Physics testing requires the measurement of various Control Element Assemblies (CEAs) bank reactivity worths at low reactor power levels. This evolution is currently performed under Technical Specification Special Test Exception 3.10.3. However, that specification test exception was written for partial RCS flow conditions which are not applicable during normal physics testing. The more appropriate test exception to be utilized is 3/4.10.2; however, this test exception references footnote (C) to Table 3.3-1. This footnote allows manual bypass of the trip below 5 percent of rated thermal power only for conduct of special test exception 3/4.10.3. Since the Core Protection Calculators (CPCs) are programmed to trip the reactor on abnormal CEA configurations, it is necessary to raise the Plant Protection System (PPS) 10^{-4} percent power bistable to 5 percent power. This bistable prevents CPC generated trips from causing a reactor trip below the PPS bistable setpoint. The first part of the proposed change would reference Tables 2.2-1, Reactor Protective Instrumentation Trip Setpoint Limits, and 3.3-1, Reactor Protective Instrumentation, in the body of Special Test Exception 3.10.2, thus allowing this exception to be used for physics testing. This part of the change would also modify footnote (5) of Table 2.2-1 to indicate that the bypass setpoint may be changed during testing pursuant to Special Test Exception 3.10.2. This footnote affects the Local Power Density-High and DNBR-Low entries in the table. Finally, this part of the proposed change would also affect footnote (C) of Table 3.3-1 to indicate that, during testing pursuant to Special Test Exception 3.10.2 (as well as 3.10.3), the trip may be manually bypassed below 5 percent of RATED THERMAL POWER. This footnote affects the "channels to trip" column for Local Power Density-High, DNBR-Low and the Core Protection Calculator entries in this table.

The existing surveillance 4.10.2.2 requires determination of linear heat rate by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 4.2.1.3. Specification 4.2.1.3 is a surveillance requirement in Specification 3/4.2 which merely requires

that the Core Operating Limit Supervisory System (COLSS) Margin Alarm be verified to actuate at the specified interval. The second part of the proposed Technical Specification change would change surveillance 4.10.2.2 to reference surveillance 4.2.1.2. This surveillance specifies conditions within which linear heat rate is to be determined and is, therefore, the appropriate surveillance to reference.

Table 3.3-1 currently does not exempt Control Element Assembly Calculators (CEACs) from Specification 3.0.4. Action 6 of the table requires that, 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 CEAs in its group. After 7 days the action allows continued operation provided that certain conditions are met. These conditions are the same as those conditions required for operation when both CEACs are inoperable. Under the conditions of Action 6b, operation may continue indefinitely. CEAC inoperability has in itself the ability to delay plant startup since the specification is not 3.0.4 exempt. An exemption for CEACs from Specification 3.0.4 would preclude this possible startup delay. The third part of the proposed change would therefore modify Action 6 of Table 3.3-1 by inserting a Specification 3.0.4 exemption for inoperability of one or both CEACs (thus allowing plant MODE changes to be made).

Safety Analysis:

The proposed changes 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 this proposed change involve a significant increase in the probability or consequences of any accident previously evaluated?

   Response: No

   Physics testing requires the measurement of CEA bank reactivity worth measurements. The CPCs are programmed to trip on abnormal CEA configurations which are routinely generated during this testing. It is, therefore, necessary to be able to raise the bistable setpoint for Local Power Density-High and DNBR-Low plant trips during physics testing. Rod testing is conducted as a normal part of approved physics testing and the current test method utilizes the raising of the bistable trip setpoint during the testing. The testing is currently performed under Special Test Exception 3.10.3 to allow raising the bistable setpoint. However, the more appropriate test exception is 3.10.2. The proposed change administratively alters Special Test Exception 3.10.2 to allow performance of testing as is now accomplished under Exception 3.10.3. Since the proposed change does not alter the physics test program, there is no change in the probability or consequences of a previously analyzed accident.

The second portion of the proposed change corrects an error in the reference to the appropriate surveillance requirement. Currently the user is referred to a 31 day surveillance on the COLSS Margin Alarm when he should be directed to the surveillance addressing Linear Heat Rate. This change corrects an error and does not affect the probability or consequences of a previously analyzed accident.

The third portion of the proposed change would add an exemption to Specification 3.0.4 for one or both CEACs inoperable. In the case of a single CEAC inoperable, the proposed conditions under which the plant would be operated (penalty factors inserted) are the same as for both CEACs out of service. This condition is already addressed in the Technical Specifications and is considered acceptable. CEA position information is only required if actual rod position information is needed to gain additional operating margin at power. In the case of both CEACs inoperable, the CPCs receive a penalty factor which ignores CEAC input. By assuming the worst CEA configuration (implementing Action 6b), additional operating margin is not required. This mode of operation is identical to the CE analog plants which normally run with no CEA position information. Therefore, operation of the facility in accordance with this proposed change is not different from the case of CE analog plants and does not increase the probability or consequences of a previously analyzed accident.

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; therefore, the proposed change does 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 the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The proposed change to Special Test Exception 3.10.2 allows resetting of the plant trip bistable from 10^{-4}% to 5% power during rod testing. The testing is part of an approved physics test program. This change does not affect a safety margin in that there are no changes to the physics test program or the conditions under which it is conducted.

   The proposed change to 4.10.2.2 simply ensures that the intended surveillance is performed by correcting the surveillance to reference the appropriate action. This correction of an error does not affect any margin of safety.

Allowing a Specification 3.0.4 exemption in the case of one or both CEACs being inoperable does not reduce a safety margin since sufficient margin is reserved in the CPCs to compensate for this mode of operation. Since appropriate penalty factors will be inserted into the CPCs and the worst case penalty factors are bounded by the Action 6.b statement and the margin associated with CE analog plants, the proposed change does not involve a significant 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 (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. An exemption to Specification 3.0.4 is normally authorized in those cases.

The proposed change to Special Test Exception 3.10.2 corrects an error in the Technical Specifications in that it allows performance of physics testing under the appropriate exception. The change to Surveillance 4.10.2.2 also corrects an error by directing the operator to the appropriate follow-on action. Specification 3.0.4 precludes entry into an operational MODE or other specified condition unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. An exemption to Specification 3.0.4 allows plant startup in those cases where a specified system or component is in an inoperable condition. An exemption to Specification 3.0.4 is normally authorized in those cases where system or component inoperability would be allowed indefinitely under the provisions of the ACTION requirements. Operation of the plant given inoperability of both CEACs is currently allowed to continue indefinitely under the conditions specified in Action 6.b. An exemption to Specification 3.0.4 should be allowed under these conditions consistent with the generic intent of the exemption. The proposed change corrects errors in Special Test Exception 3.10.2 and in Surveillance 4.10.2.2 and achieves consistency within the Technical Specifications relative to the applicability of the 3.0.4 exemption and is, therefore, similar to Example (i).

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; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; 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.

BRD:8522F

ATTACHMENT A

UNIT 2: EXISTING TECHNICAL SPECIFICATIONS

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grids is 1.31.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7Λ
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	7Λ
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2	2#, 3#
				3*, 4*, 5*	7Λ
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	7Λ
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY wthin 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

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ATTACHMENT B

UNIT 2: PROPOSED TECHNICAL SPECIFICATIONS

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grids is 1.31. The bypass setpoint may be changed during testing pursuant to Special Test Exception 3.10.2.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)(e)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)(e)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2	2#, 3#
				3*, 4*, 5*	7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	7A
14. Core Protection Calculators	4	2(c)(d)(e)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6#, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed then THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.2 or 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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 CEAs in its group. After 7 days, operation may continue provided that Action 6.b is met.* If the exemption to Specification 3.0.4 is used, Action 6.b must be 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.

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

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at lest once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

ATTACHMENT C

UNIT 3: EXISTING TECHNICAL SPECIFICATIONS

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grid is 1.31.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2	2#, 3#
				3*, 4*, 5*	7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	7A
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

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 coninue provided that:*

1. Within 1 hour the DNBR margin reqired 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.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

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ATTACHMENT D

UNIT 3: PROPOSED TECHNICAL SPECIFICATIONS

TABLE 2.2-1 (Continued)
REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. The approved DNBR limit accounting for use of HID-2 grids is 1.31. The bypass setpoint may be changed during testing pursuant to Special Test Exception 3.10.2.

(6) DN RATE is the maximum decrease rate of the trip setpoint.

FLOOR is the minimum value of the trip setpoint.

STEP is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

(7) Acceleration, horizontal/vertical, g.

(8) Setpoint may be altered to disable trip function during testing pursuant to Specification 3.10.3.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2	1 set of 2	2 sets of 2	1, 2	1
	2 sets of 2	1 set of 2	2 sets of 2	3*, 4*, 5*	7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level - High					
a. Startup and Operating	4	2(a)(d)	3	1, 2	2#, 3#
	4	2	3	3*, 4*, 5*	7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)(e)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)(e)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2	2#, 3#
				3*, 4*, 5*	7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2	5
				3*, 4*, 5*	7A
14. Core Protection Calculators	4	2(c)(d)(e)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6#, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 400 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed then THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.2 or 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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 CEAs in its group. After 7 days, operation may continue provided that Action 6.b is met.* If the exemption to Specification 3.0.4 is used, Action 6.b must be 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.

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

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at lest once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.2 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, footnote C of Table 3.3-1 and footnote 5 of Table 2.2-1 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGE NPF-10/15-234

This is a request to revise Technical Specification 3/4.5.1, "Safety Injection Tanks".

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 3/4.5.1, "Safety Injection Tanks". The existing Limiting Condition for Operation (LCO) 3.5.1.d requires that each reactor coolant system safety injection tank be OPERABLE with a nitrogen cover-pressure of between 600 and 625 psig. This requirement in conjunction with other requirements of the LCO ensures that a sufficient volume of borated water will be immediately forced into the reactor core through the cold legs of the Reactor Coolant System (RCS) in the event that the RCS pressure falls below the pressure of the safety injection tanks (SITs). This initial surge of water into the core provides the initial cooling mechanism during large pipe ruptures within the reactor coolant pressure boundary.

The proposed change revises the required upper limit of the nitrogen cover-pressure from 625 psig to 640 psig. This change is required to prevent possible violation of the pressure limit in the other SITs due to inleakage from the common fill header when one of the tanks is being filled to maintain its pressure within limits. In addition, the proposed change revises the units of pressure from pounds per square inch gauge (psig) to pounds per square inch absolute (psia). This change is required to make the units of measurement consistent with other units on the control room panel. Finally, the proposed change also deletes the Unit-3 Cycle-2 specific lower SIT boron concentration requirement from the Limiting Condition for 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 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 in SIT nitrogen pressure is on the upper limit of the pressure band. Since LOCA analyses are based on the lower limit of the band (600 psig) which is unchanged, there will be no effect on reflood rate and hence no effect on core LOCA limits.

   The proposed change results in an increase of about 2-1/2% in the mass of nitrogen cover gas in the SIT, when at the new upper limit pressure of 640 psig. Thus, a LOCA using the new upper band pressure as an initial condition would have a slight increase in nitrogen flow (after injection of SIT water) in the injection nozzles over a slightly longer time. This increase in nitrogen flow could produce a slightly higher resistance to the flow of reflood-generated steam in the cold leg nozzles. The effect of this slightly higher resistance on reflood rate would be negligible.

   Therefore, operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of accidents 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 result in any change in plant hardware, plant operation, or operating procedures. Therefore, operation of the facility in accordance with 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

   There is no reduction in a margin of safety since the SITs with a higher upper limit on cover gas pressure will provide the same protection against a postulated accident as before. Therefore, operation of the facility in accordance with this proposed change will not involve a significant 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 (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 (vi) relates to a change which either may result in some increase in 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 refinement of a previously used calculation model or design method. Acceptance criteria of SRP 15.6.5 require that ECCS equipment refill the vessel in a timely manner for a LOCA and provide adequate core cooling during a LOCA. The SITs will perform both these functions as before. Therefore, the proposed change to raise the maximum SIT pressure from 625 psig to 640 psig is similar to Example (vi) in that the proposed change may result in an insignificant increase in the consequences of a previously analyzed accident, but where the results of the change are clearly within all acceptable criteria with respect to the SITs specified in the SRP Section 15.6.5. The proposed change to revise the unit of cover gas pressure from psig to psia is similar to Example (i) in that it is purely an administrative change that revises nomenclature, (i.e., units of measure). The proposed change to eliminate the Unit-3 Cycle-2 specific, lower SIT boron concentration from the Limiting Condition for Operation is administrative in nature and similar to Example (i) since Unit-3 Cycle-2 operation has been completed.

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; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; 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.

BRD:8512F

ATTACHMENT A

Existing Technical Specification, Unit 2

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.

---

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

ATTACHMENT B

Proposed Technical Specification, Unit 2

3/4.1 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 2500 ppm of boron, and

d. A nitrogen cover-pressure of between 615 and 655 psia.

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 ne 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.

ATTACHMENT C

Existing Technical Specification, Unit 3

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 (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.

ATTACHMENT D

Proposed Technical Specification, Unit 3

3/4.1 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 2500 ppm of boron, and

d. A nitrogen cover-pressure of between 615 and 655 psia.

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 ne 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.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-235

This is a request to revise Technical Specification (TS) 3/4.1.1.1, "Boration Control - Shutdown Margin Tavg > 200°F," TS 3/4/.1.1.2, "Shutdown Margin - Tavg ≤ 200°F," TS 3/4.1.2.7, "Borated Water Source - Shutdown," TS 3/4.1.2.8, "Borated Water Sources - Operating," TS 3/4.5.1, "Safety Injection Tanks," TS 3/4.5.4, "Refueling Water Storage Tank," TS 3/4.6.2.2, "Recirculation Flow pH Control," TS 3/4.9.1, "Refueling Operations - Boron Concentration," TS 3/4.10.1, "Special Test Exceptions - Shutdown Margin," 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 Specification (TS) 3/4.1.1.1, "Boration Control - Shutdown Margin Tavg > 200°F," TS 3/4/.1.1.2, "Shutdown Margin - Tavg ≤ 200°F," TS 3/4.1.2.7, "Borated Water Source - Shutdown," TS 3/4.1.2.8, "Borated Water Sources - Operating," TS 3/4.5.1, "Safety Injection Tanks," TS 3/4.5.4, "Refueling Water Storage Tank," TS 3/4.6.2.2, "Recirculation Flow pH Control," TS 3/4.9.1, "Refueling Operations - Boron Concentration," TS 3/4.10.1, "Special Test Exceptions - Shutdown Margin," and Associated Bases.

These specifications define as limiting conditions for operation (LCO), volumes and concentrations of borated water to be maintained in the refueling water storage tank (RWST), the boric acid makeup (BAMU) tanks and safety injection tanks (SIT), the minimum boron concentration to be maintained in the refueling mode as well as the amount of trisodium phosphate (TSP) to be stored inside containment. These specifications also require the periodic performance of specified surveillance tests and inspections to verify that the LCO's are met, and identify compensatory actions to be taken in the event that LCO's are not met.

In normal plant operation, borated water is used to maintain reactivity control and makeup for coolant contraction during plant cooldown. During postulated accidents, borated water would be injected into the reactor coolant system (RCS) to maintain RCS inventory and ensure the reactor is subcritical. In a postulated loss of coolant accident (LOCA) the reactor core is reflooded by borated water injected by the SIT's, the safety injection pumps taking suction from the RWST, and the charging pumps taking suction from the BAMU tanks. The RWST also provides a source of borated water for the containment spray system which suppresses containment pressure during postulated LOCA's

and steamline break accidents. During a LOCA, borated water spilled from the RCS accumulates along with containment spray water in the containment sump. After the RWST inventory is exhausted, the emergency core cooling and containment spray systems continue recirculating borated water from the containment sump.

Trisodium phosphate stored in baskets in the containment sump dissolve in the accumulated water raising its pH. This minimizes stress corrosion cracking of metals inside containment.

The proposed change will revise borated water source concentration and volume requirements consistent with the assumptions and results of analysis supporting extended fuel cycles. Extended fuel cycles have necessitated an increase in the refueling boric acid concentration. The proposed change also makes a corresponding increase in TSP requirements to ensure the ability to neutralize the increased amount of boric acid.

The proposed change: 1) increases the minimum refueling concentration and RWST lower limit on boron concentration from 1720 to 2350 ppm; 2) increases the RWST and SIT upper limits on concentration from 2500 to 2800 ppm; 3) revises Figure 3.1-1 which specifies the concentration and volume of boric acid to be maintained in the BAMU tanks during operation based on RWST concentration to correspond to the proposed RWST concentration ranges; 4) decreases the volume of borated water required in either the BAMU tanks or RWST for when the plant is in cold shutdown or refueling from 5150 to 4150 gallons; 5) increases the lower limit on SIT concentration from 1720 to 1850 ppm; and 6) increases the amount of TSP required in containment from 15,400 lbs to 17,461 lbs.

1) Increase in minimum refueling mode and RWST boron concentration.

The proposed change increases the minimum refueling mode and RWST boron concentration from 1720 to 2350 ppm. The proposed 2350 ppm lower limit is based on the refueling concentration assumed as an initial condition for the Mode 6 (Refueling) boron dilution event analysis. An assumed initial refueling boron concentration of 2300 ppm will preserve a minimum of 60 minutes to criticality in Mode 6 with the RCS at mid-loop and 3 charging pumps running. A 50 ppm uncertainty allowance is added to the assumed value to arrive at the proposed 2350 ppm refueling concentration. The proposed change also revises to "2350 ppm" all references to "1720 ppm" in TS 3/4.1.1.1, "Boration Control - Shutdown Margin - Tavg > 200°F," TS 3/4.1.1.2, "Shutdown Margin - Tavg ≤ 200°F," TS 3/4.1.2.7, "Borated Water Source - Shutdown," TS 3/4.1.2.8, "Borated Water Source - Operating," TS 3/4.5.4, "Refueling Water Storage Tank," TS 3/4.9.1, "Refueling Operations - Boron Concentration," and TS 3/4.10.1, "Special Test Exceptions - Shutdown Margin." Similar references in the associated bases sections are also revised.

2) Increase in RWST and SIT boron concentration upper limit.

The proposed change increases the upper limit on boron concentration from 2500 ppm to 2800 ppm boron. This increase is required to preserve an adequate operating range of concentrations with the increase in lower limits described above. An upper limit on concentration is specified in ensure 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. Although the proposed change would result in a higher post-LOCA boric acid concentration (a maximum of 2812 ppm boron), the concentration remains less than the solubility limit for post-LOCA conditions. Accordingly, the proposed change revises references to 2500 ppm upper limits to 2800 ppm in TS 3/4.1.2.8, "Borated Water Sources - Operating," TS 3/4.5.1, "Safety Injection Tanks," TS 3/4.5.4, "Refueling Water Storage Tank." The proposed change also revises Surveillance Requirement 4.6.2.2.b of TS 3/4.6.2.2, "Recirculation Flow pH Control" which uses an aggregate post LOCA concentration to test trisodium phosphate stored in containment. The proposed change increases this concentration from 2482 to 2812 ppm boron.

3) Revisions to Figure 3.1-1

Figure 3.1-1 define the volumes and concentrations of boric acid to be maintained in the BAMU tanks corresponding to various RWST concentrations. Currently Figure 3.1-1 incorporates four curves which represent the minimum boric acid volume as a function of concentration required from BAMU tanks for RWST concentrations of 1720, 2000, 2300, and 2500 ppm. The proposed change revises Figure 3.1-1 to include four curves for RWST concentrations of 2350, 2500, 2650 and 2800 ppm, the new range of RWST concentrations described above. The new curves were generated using the same methodology as was used for the existing curves which is documented in CEN-316, "Boric Acid Makeup Tank Concentration Reduction Effort" and was previously reviewed and approved for San Onofre Nuclear Generating Station Units 2 and 3, in conjunction with the issuance of Amendments 43 and 32 respectively. The proposed figure ensures that sufficient inventory is maintained in the BAMU tanks to support a natural circulation cooldown while maintaining 5.15% shutdown margin, to the point where the cooldown can continue to cold shutdown with inventory and boron addition from the RWST, assuming a loss of offsite power, a limiting single failure and that the letdown line is unavailable. The minimum BAMU tank inventory specified in Figure 3.1-1, 4151 gallons, is sufficient to support credit taken for charging pump operation in a postulated small break LOCA.

4) Decrease in cold shutdown BAMU tank/RWST inventory requirement.

The proposed change reduces the BAMU tank or RWST inventory requirements specified in TS 3/4.3.1.2.7, "Borated Water Source - Shutdown" from 5150 to 4150 gallons. The volume and concentration specified is based on maintaining a minimum of 3.0% shutdown margin after xenon decay during cooldown from 200°F to 140°F. 4150 gallons are required to makeup for

coolant contraction during the cooldown. With the increase in minimum boron concentration from 1720 to 2350 ppm, this volume also contains in excess of the boron required to maintain 3.0% shutdown margin.

5) Increase in SIT lower concentration limit.

TS 3/4.5.1, "Safety Injection Tanks," specifies a minimum SIT concentration of 1720 ppm boron. The proposed change increases this to 1850 ppm. The lower limit on SIT concentration ensures that the reactor will be at least 1% subcritical following a large break LOCA taking no credit for control element assembly (CEA) insertion.

6) Increase in TSP requirements.

TS 3/4.6.2.2, "Recirculation Flow pH Control," requires a minimum of 15,400 lbs (256 cubic feet) TSP to be available in storage rocks in the containment. The required amount of TSP is sufficient to neutralize the maximum amount of boric acid postulated to be inside containment following a LOCA. With an increase in maximum RWST concentration to 2800 ppm, an increase in TSP to 17,461 lbs (291 cubic feet) is required. The proposed change correspondingly increases the amount of TSP specified in surveillance requirement 4.6.2.2.b from 3.00 grams to 3.43 grams.

The proposed change also removes from Bases Sections 3/4.1.2, "Boration Systems" and 3/4.5.4, "Refueling Water Storage Tank" references to sodium hydroxide (NaOH) for pH control. These changes to the bases were inadvertently overlooked in conjunction with Amendments 51 and 40 for Units 2 and 3, respectively, which approved deletion of the NaOH iodine removal system.

Safety Analysis

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 revises borated water source concentration and volume requirements. Borated water sources are credited in the LOCA and steamline break analysis, establish initial conditions for refueling mode boron dilution event and maintain reactivity control and makeup for coolant contraction during plant cooldown.

The LOCA assumes that borated water is injected into RCS from the SIT's and by high and low pressure safety injection (HPSI and LPSI) pumps taking suction from the RWST. The proposed change requires a minimum

concentration of 1850 ppm boron be maintained in the SIT's and 2350 ppm in the RWST. These concentrations are sufficient to maintain the reactor subcritical during a LOCA assuming no CEA's are inserted. The small break LOCA relies on flow from a charging pump to augment flow from the HPSI and LPSI pumps. On a safety injection actuation signal (SIAS) the charging pumps are aligned to take suction from the BAMU tanks. The proposed change reduces the minimum inventory for the BAMU tanks. However, sufficient inventory is maintained to support charging flow during the course of a small break LOCA. The inventory requirements for the SIT's and RWST are unaffected by the proposed change.

The proposed change increases the upper limit on SIT and RWST concentration. This increases the total amount of boric acid injected during a LOCA from the RWST and SIT's. Although the proposed change would result in a higher post-LOCA boric acid concentration, the concentration remains less than the solubility limit for post-LOCA conditions.

Operation of the safety injection system is credited in the steamline break analysis for the addition of negative reactivity. The steamline break analysis is bounded by the Cycle 3 analysis which assumes that a minimum concentration of 1720 ppm boron from the RWST is injected. Thus the analysis is not adversely affected by the proposed increase in minimum RWST concentration.

The Mode 6 boron dilution event analysis currently demonstrates a minimum of 60 minutes to criticality with one charging pump operating. The proposed change would increase the minimum refueling concentration to 2350 ppm. Reanalysis of the Mode 6 boron dilution event at mid-loop with this initial concentration demonstrates a minimum of 60 minutes to criticality with all three pumps operating. Thus the proposed change does not affect the results of the Mode 6 boron dilution event.

Borated water from the BAMU tanks and RWST are required for reactivity control and makeup during cooldown. The limiting cooldown scenario is that analyzed to demonstrate conformance with Branch Technical Position RSB 5-1. This scenario assumes that letdown is unavailable, a loss of offsite power, and the limiting single failure. Sufficient borated water must be available to maintain shutdown margin and makeup for RCS coolant contraction during cooldown. The proposed Figure 3.1-1 was developed from the same methodology as was used for the current Figure 3.1-1 approved by Amendment Nos. 43 and 32 and described in CEN-316. The higher RWST concentrations proposed result in lower BAMU tank inventory requirements.

Assuming that the RWST outlet valve in the gravity feed path to the charging pump suctions fails closed, from the standpoint of RCS coolant contraction 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 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 evaluated accident.

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 LOCA, steamline break, boron dilution event and safe shutdown analysis. 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 considerations 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 for a nuclear power reactor resulting from a core reloading if no fuel assemblies are significantly different from those found previously acceptable to the NRC. This assumes that no significant changes are made to the acceptance criteria for the Technical Specification and that the analytical methods used to demonstrate conformance with Technical Specifications and regulations are not significantly changed and that the NRC has previously found such methods acceptable.

The proposed change revises borated water source and trisodium phosphate requirements as a result of increases in critical boron concentration related to extended fuel cycles. The proposed changes to specified values are either bounded by analysis presented in the FSAR or were generated using previously reviewed and approved methodology.

Specifically, the proposed increase in minimum RWST and refueling boron concentration preserves a minimum of 60 minutes to criticality for the refueling mode boron dilution event analysis, using the previously reviewed and approved analytical methodology presented in the FSAR. The proposed upper limits on concentration for the RWST and SIT continue to ensure the boric acid does not precipitate in the core during the long term cooling phase following a LOCA. The proposed revision to BAMU tank inventory requirements were generated using the same analytical methodology as the existing requirements which was approved by Unit 2 Amendment 43 and Unit 3 Amendment 32.

The proposed reduction in BAMU tank or RWST inventory from 5150 to 4150 gallons with the plant in cold shutdown or refueling uses the same methodology as was used for the current value, taking into account the higher proposed minimum concentration. The proposed value maintains in excess of 3.0% shutdown margin during a cooldown from 200°F to 140°F.

The proposed change would increase the minimum SIT boron concentration from 1720 to 1850 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. In the large break LOCA analysis, conservatively, no credit is taken for control rod insertion. The proposed change ensures that the reactor would be maintained subcritical during a LOCA and therefore continues to meet this acceptance criteria.

The proposed increase in TSP from 15,400 to 17,461 lbs compensates for the increase in RWST and SIT maximum boric acid concentration. SRP Section 6.1.1, "Engineered Safety Feature Materials," requires that the composition of containment spray and core cooling water be controlled to ensure a minimum pH of 7.0 following a LOCA to inhibit initiation of stress corrosion cracking. The proposed increase in TSP requirements continues to meet this SRP criterion.

The proposed changes are related to a core reloading for extended fuel cycles, are generated by previously approved analytical methodology or are bounded by previously reviewed analysis, and preserve existing acceptance criteria. Therefore, the proposed change is similar to Example (iii) and does not involve significant hazards consideration.

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:8518F

ATTACHMENT "A"

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5.15% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.15% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

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

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.
   7. Whenever the reactor coolant level is below the hot leg centerline, one and only one charging pump shall be operable; by verifying that power is removed from the remaining charging pumps.

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

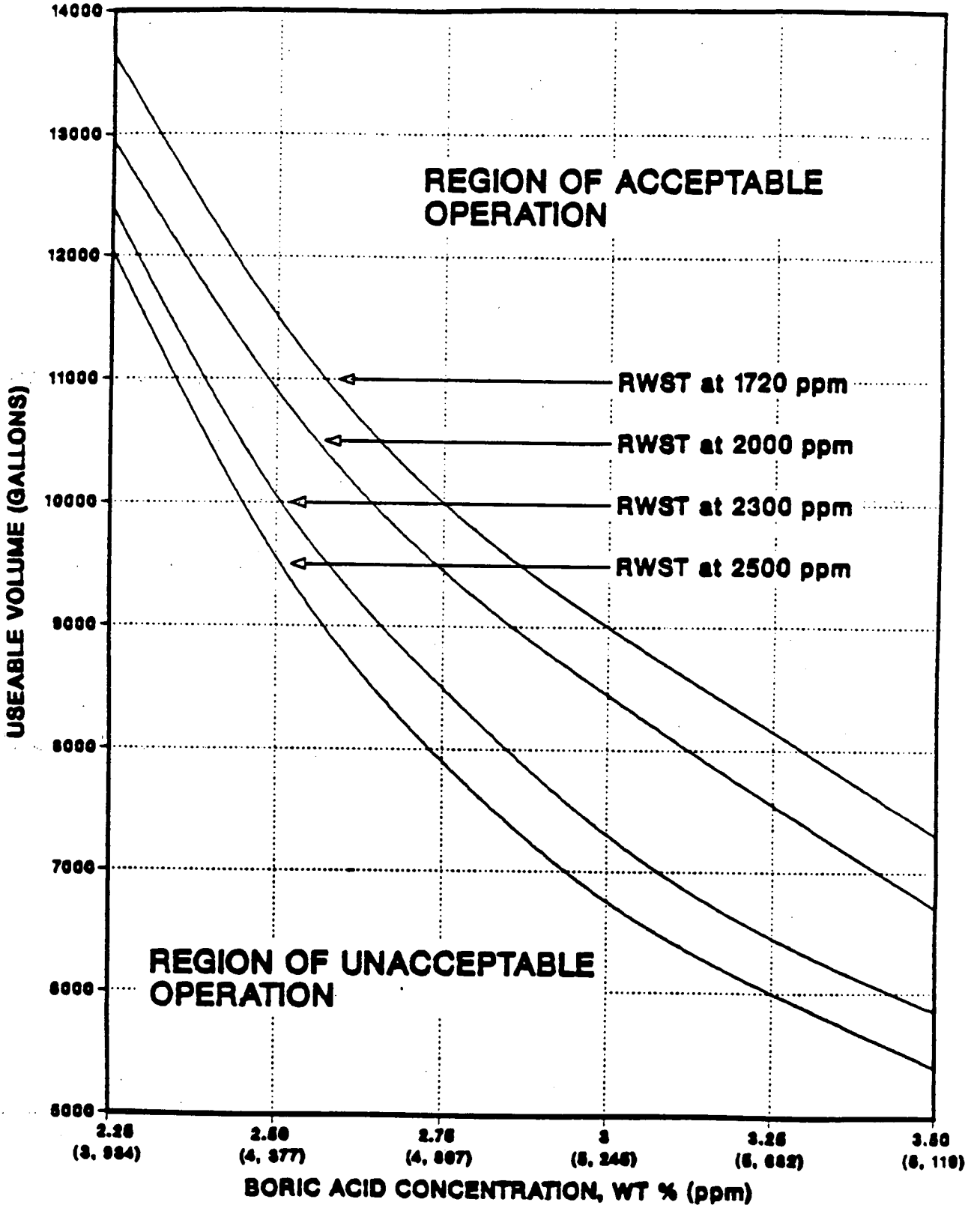

Figure 3.1-1

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 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 703 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.

RECIRCULATION FLOW PH CONTROL

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation flow pH control system shall be operable with a minimum of 15,400 lbs. (256 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in the storage racks in the containment.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

With less than the required amount of trisodium phosphate available, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The recirculation flow pH control system shall be demonstrated operable during each refueling outage by:

a. Visually verifying that the TSP storage racks have maintained their integrity and the TSP containers contain a minimum of 15,400 lbs. (256 cu. ft.) of TSP (w/12 hydrates) or equivalent.

b. Verifying that when a sample of less than 3.00 grams of trisodium phosphate (w/12 hydrates) or equivalent, selected at random from one of the storage racks inside of containment, is submerged, without agitation, in at least 1 litre of 120 ± 10 degrees-F borated demineralized water borated to at least 2482 ppm boron, allowed to stand for 4 hours, then decanted and mixed, the pH of the solution is greater than or equal to 7.0.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

a. Either a K_{eff} of 0.95 or less,

b. A boron concentration of greater than or equal to 1720 ppm,

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 1720 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

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

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

*Operation in MODE 3 shall be limited to 6 consecutive hours.

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 c consistency the minimum required volume of 362,800 gallons above ECCS suction connectior 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 1% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires 5150 gallons of 1720 ppm borated water from either the refueling water tank or boric acid solution from a boric acid makeup tank.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

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

The limits on water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable, CEA to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

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ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The NaOH added to the Containment Spray, via the Spray Chemical Addition pumps, minimizes the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The NaOH additive results in post-LOCA sump pH of between 8.0 and 10.0 at the end of the NaOH injection period.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. The limit on maximum boron concentration is to ensure that boron does not precipitable in the core following LOCA. The limit on RWST solution temperature is to ensure that the assumptions used in the LOCA analyses remain valid.

REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification). The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the mimumum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The limits on water volume and boron concentration of the RWST also ensure that the solution recirculated within containment after a LOCA has a pH value between 8.0 and 10.0 at the end of the NaOH injection period. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 1720 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.9.4 CONTAINMENT PENETRATIONS

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

3/4.9.5 COMMUNICATIONS

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

ATTACHMENT "B"

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5.15% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.15% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

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

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.
   7. Whenever the reactor coolant level is below the hot leg centerline, one and only one charging pump shall be operable; by verifying that power is removed from the remaining charging pumps.

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 2350 ppm and a minimum borated water volume of 4150 gallons, or

b. The refueling water storage tanks with:

1. A minimum borated water volume of 4150 gallons above the ECCS suction connection,

2. A minimum boron concentration of 2350 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:

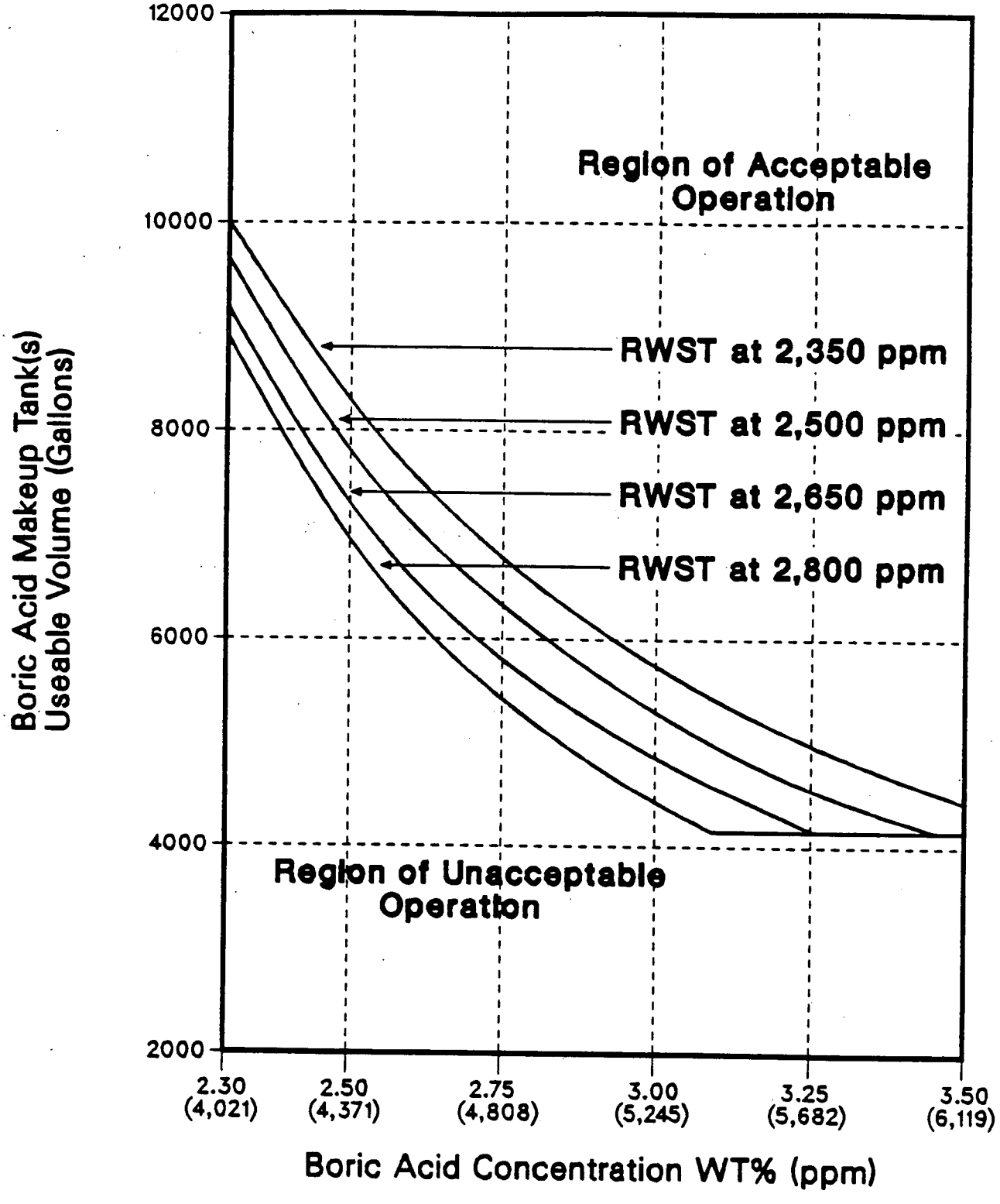
a. At least once per 7 days by:

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
MINIMUM STORED BORIC ACID VOLUME AS A FUNCTION OF CONCENTRATION (Gallons)



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 2350 and 2800 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 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 1850 and 2800 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 723 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 2350 and 2800 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.

RECIRCULATION FLOW PH CONTROL

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation flow pH control system shall be operable with a minimum of 17,461 lbs. (291 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in the storage racks in the containment.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

With less than the required amount of trisodium phosphate available, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The recirculation flow pH control system shall be demonstrated operable during each refueling outage by:

a. Visually verifying that the TSP storage racks have maintained their integrity and the TSP containers contain a minimum of 17,461 lbs. (291 cu. ft.) of TSP (w/12 hydrates) or equivalent.

b. Verifying that when a sample of less than 3.43 grams of trisodium phosphate (w/12 hydrates) or equivalent, selected at random from one of the storage racks inside of containment, is submerged, without agitation, in at least 1 litre of 120 ± 10 degrees-F borated demineralized water borated to at least 2812 ppm boron, allowed to stand for 4 hours, then decanted and mixed, the pH of the solution is greater than or equal to 7.0.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

a. Either a K_{eff} of 0.95 or less,

b. A boron concentration of greater than or equal to 2350 ppm,

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2350 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

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

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

*Operation in MODE 3 shall be limited to 6 consecutive hours.

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 2350 ppm borated water from the refueling water tank or approximately 26,000 gallons of 2350 ppm borated water from the refueling water tank alone. However, for the purpose c consistency the minimum required volume of 362,800 gallons above ECCS suction connectior 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 4150 gallons of 2350 ppm borated water from either the refueling water tank or boric acid solution from a boric acid makeup tank.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

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

The limits on water volume and boron concentration of the RWST also ensure a pH value of greater than 7.0 for the solution recirculated within containment after a LOCA. This pH minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable, CEA to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. The limit on maximum boron concentration is to ensure that boron does not precipitable in the core following LOCA. The limit on RWST solution temperature is to ensure that the assumptions used in the LOCA analyses remain valid.

REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification). The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the mimumum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The limits on water volume and boron concentration of the RWST also ensure that the solution recirculated within containment after a LOCA has a pH value of greater than 7.0. This pH minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 2350 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.9.4 CONTAINMENT PENETRATIONS

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

3/4.9.5 COMMUNICATIONS

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

ATTACHMENT "C"

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5.15% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.15% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

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

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.
   7. Whenever the reactor coolant level is below the hot leg centerline, one and only one charging pump shall be operable; by verifying that power is removed from the remaining charging pumps.

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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:

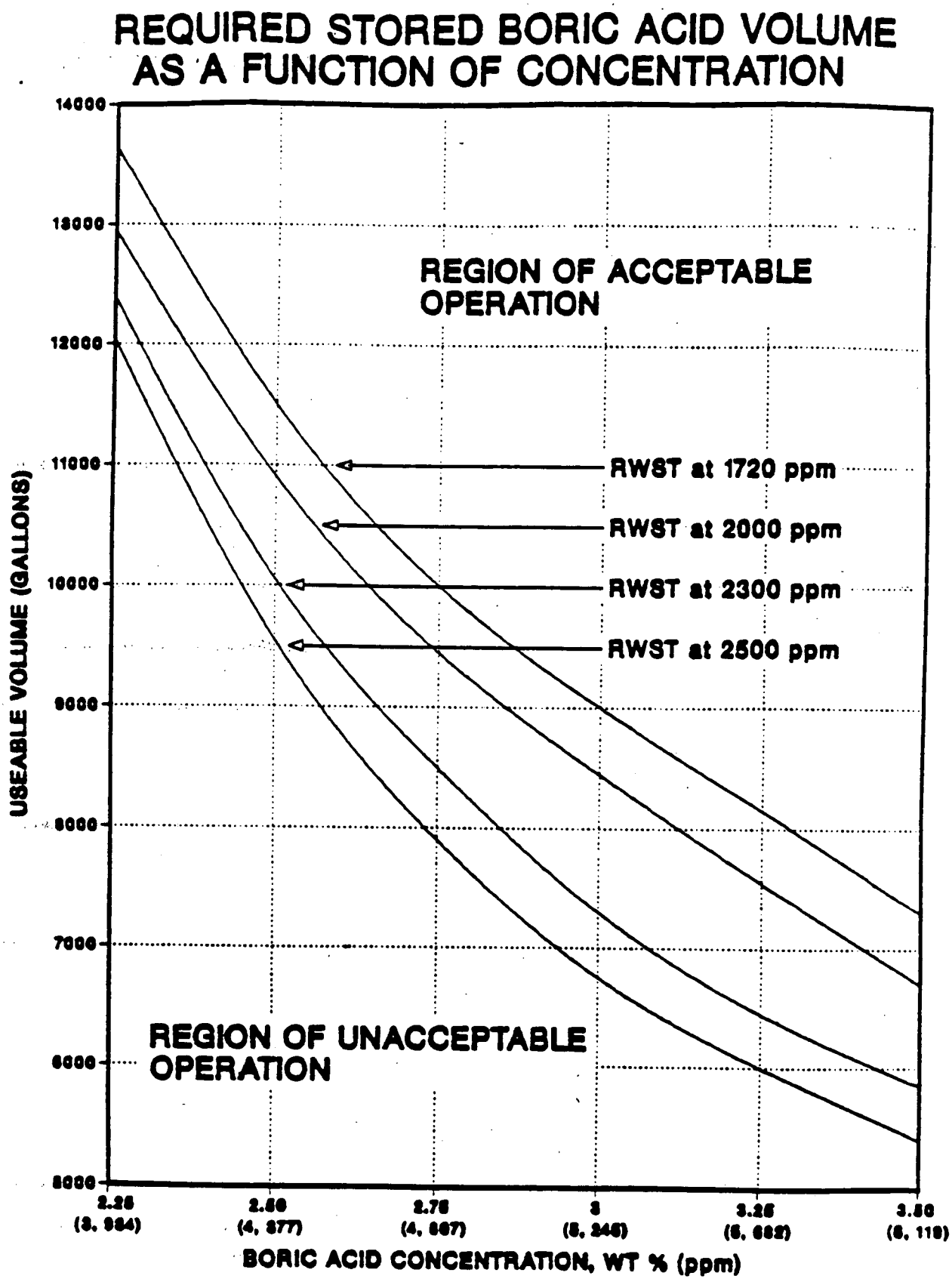
a. At least once per 7 days by:

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
14000
13000
12000
11000
10000
9000
8000
7000
6000
5000
USEABLE VOLUME (GALLONS)
REGION OF ACCEPTABLE
OPERATION
RWST at 1720 ppm
RWST at 2000 ppm
RWST at 2300 ppm
RWST at 2500 ppm
REGION OF UNACCEPTABLE
OPERATION
2.25
(3, 984)
2.50
(4, 377)
2.75
(4, 667)
3
(5, 246)
3.25
(5, 662)
3.50
(6, 118)
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°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 EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.2 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

4.5.2 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.

RECIRCULATION FLOW - PH CONTROL

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation flow pH control system shall be operable with a minimum of 15,400 lbs. (256 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in the storage racks in the containment.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

With less than the required amount of trisodium phosphate available, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The recirculation flow pH control system shall be demonstrated operable during each refueling outage by:

a. Visually verifying that the TSP storage racks have maintained their integrity and the TSP containers contain a minimum of 15,400 lbs. (256 cu. ft.) of TSP (w/12 hydrates) or equivalent.

b. Verifying that when a sample of less than 3.00 grams of trisodium phosphate (w/12 hydrates) or equivalent, selected at random from one of the storage racks inside of containment, is submerged, without agitation, in at least 1 litre of 120 ± 10 degrees-F borated demineralized water borated to at least 2482 ppm boron, allowed to stand for 4 hours, then decanted and mixed, the pH of the solution is greater than or equal to 7.0.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

a. Either a K_{eff} of 0.95 or less,

b. A boron concentration of greater than or equal to 1720 ppm,

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 1720 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

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

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

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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, 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 consistency the minimum required volume of 362,800 gallons above ECCS suction connectio 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 1% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires ≥ 5150 gallons of 1720 ppm borated water from either the refueling water tank or boric acid solution from a boric acid makeup tank.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

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

The limits on water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time-dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

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ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The NaOH added to the Containment Spray, via the Spray Chemical Addition pumps, minimizes the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The NaOH additive results in post-LOCA sump pH of between 8.0 and 10.0 at the end of the NaOH injection period.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. The limit on maximum boron concentration is to ensure that boron does not precipitate in the core following LOCA. The limit on RWST solution temperature is to ensure that the assumptions used in the LOCA analyses remain valid.

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REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the minimum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The limits on water volume and boron concentration of the RWST also ensure that the solution recirculated within containment after a LOCA has a pH value between 8.0 and 10.0 at the end of the NaOH injection period. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

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3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 1720 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

3/4.9.5 COMMUNICATIONS

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

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ATTACHMENT "D"

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 5.15% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.15% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

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* See Special Test Exception 3.10.1.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 3.0% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 3.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 3.0% delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

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

   1. Reactor coolant system boron concentration,
   2. CEA position,
   3. Reactor coolant system average temperature,
   4. Fuel burnup based on gross thermal energy generation,
   5. Xenon concentration, and
   6. Samarium concentration.
   7. Whenever the reactor coolant level is below the hot leg centerline, one and only one charging pump shall be operable; by verifying that power is removed from the remaining charging pumps.

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 2350 ppm and a minimum borated water volume of 4150 gallons, or

b. The refueling water storage tanks with:

1. A minimum borated water volume of 4150 gallons above the ECCS suction connection,

2. A minimum boron concentration of 2350 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:

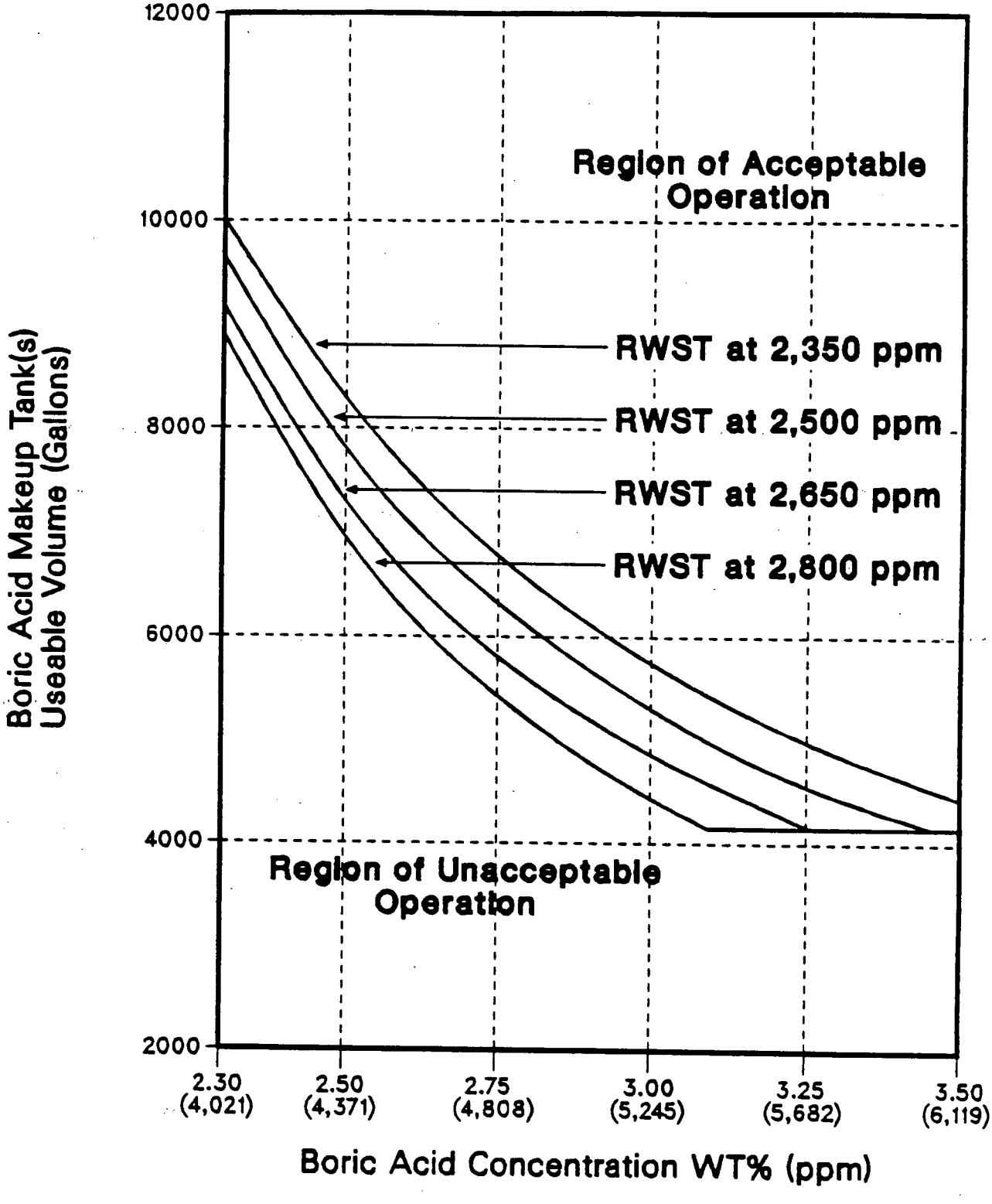
a. At least once per 7 days by:

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
MINIMUM STORED BORIC ACID VOLUME AS A FUNCTION OF CONCENTRATION (Gallons)



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 2350 and 2800 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 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 1850 and 2800 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.

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 2350 and 2800 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.

RECIRCULATION FLOW - PH CONTROL

LIMITING CONDITION FOR OPERATION

3.6.2.2 The recirculation flow pH control system shall be operable with a minimum of 17,461 lbs. (291 cu. ft.) of trisodium phosphate (w/12 hydrates), or equivalent, available in the storage racks in the containment.

APPLICABILITY: Modes 1, 2 and 3

ACTION:

With less than the required amount of trisodium phosphate available, restore the system to the correct amount within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The recirculation flow pH control system shall be demonstrated operable during each refueling outage by:

a. Visually verifying that the TSP storage racks have maintained their integrity and the TSP containers contain a minimum of 17,461 lbs. (291 cu. ft.) of TSP (w/12 hydrates) or equivalent.

b. Verifying that when a sample of less than 3.43 grams of trisodium phosphate (w/12 hydrates) or equivalent, selected at random from one of the storage racks inside of containment, is submerged, without agitation, in at least 1 litre of 120 ± 10 degrees-F borated demineralized water borated to at least 2812 ppm boron, allowed to stand for 4 hours, then decanted and mixed, the pH of the solution is greater than or equal to 7.0.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

a. Either a K_{eff} of 0.95 or less,

b. A boron concentration of greater than or equal to 2350 ppm,

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2350 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

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

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

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 2350 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

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*Operation in MODE 3 shall be limited to 6 consecutive hours.

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 2350 ppm borated water from the refueling water tank or approximately 26,000 gallons of 2350 ppm borated water from the refueling water tank alone. However, for the purpose consistency the minimum required volume of 362,800 gallons above ECCS suction connectio 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 ≥ 4150 gallons of 2350 ppm borated water from either the refueling water tank or boric acid solution from a boric acid makeup tank.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

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

The limits on water volume and boron concentration of the RWST also ensure a pH value of greater than 7.0 for the solution recirculated within containment after a LOCA. This pH minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time-dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. The limit on maximum boron concentration is to ensure that boron does not precipitate in the core following LOCA. The limit on RWST solution temperature is to ensure that the assumptions used in the LOCA analyses remain valid.

REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the minimum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

The limits on water volume and boron concentration of the RWST also ensure that the solution recirculated within containment after a LOCA has a pH value greater than 7.0. This pH minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1% delta K/K conservative allowance for uncertainties. Similarly, the boron concentration value of 2350 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

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

3/4.9.3 DECAY TIME

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

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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

3/4.9.5 COMMUNICATIONS

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

DESCRIPTION AND SAFETY ANALYSIS OF PROPOSED CHANGE NPF-10/15-236

This is a request to revise Technical Specification 6.9.1.14, "Hazardous Cargo Traffic Report".

Existing Technical Specification

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

Proposed Technical Specification

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

Description

The proposed change revises Technical Specification reporting requirement 6.9.1.14, "Hazardous Cargo Traffic Report". The existing reporting requirements specify that the hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years. The origin of these requirements is the San Onofre Units 2 and 3 Safety Evaluation Report (SER). In the SER, the NRC Staff concurred with SCE that the risks due to potential explosions or toxic gas releases (with the provisions for protecting against the specific toxic gases described) are acceptably low and meet the criteria described in the Standard Review Plan (SRP), Section 2.2.3. (the TGIS currently is designed to isolate the Control Room air intake upon detection of the presence of Ammonia, Chlorine, Butane and Propane). The SER conclusion was based in part on the knowledge of present sizes and frequencies of hazardous cargo shipments going past the San Onofre site. However, it was noted by the NRC Staff that significant changes over the lifetime of the plant in traffic density, transportation conditions, cargo composition, size and frequency could have a significant effect on the risk estimates. Therefore, the Staff required that the hazardous cargo traffic on I-5 and the AF&SF railway be monitored and the results periodically reported to the Staff.

The proposed change would remove the requirement to monitor and report the toxic gas cargo traffic on I-5. The requirement to monitor and report explosion hazard cargo on Interstate Route 5 and hazard cargo traffic on the AT&SF railway would remain in effect.

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

   Previously analyzed accidents include the release of toxic gases from an accident on Interstate Route 5 (I-5) which would potentially cause the San Onofre Units 2 and 3 control room envelope to become uninhabitable. In the SER, the NRC Staff concurred with SCE (based upon prior analyses) that the risks due to toxic gas releases, when coupled with the provisions for control room isolation, were acceptably low. However, the NRC Staff required that hazardous cargo traffic (explosive hazards and hazards from toxic gases) be monitored and the results periodically reported.

   SCE has performed an analysis of the toxic gas hazard shipments along I-5. This analysis demonstrates that the combined risk associated with the composite toxic gas hazard from shipments on I-5 is acceptably low and meets the acceptance criteria of Standard Review Plan (SRP) Section 2.2.3. The analysis further demonstrates that this risk remains acceptably low throughout the remainder of the San Onofre Unit 3 operating license.

   The proposed change does not alter the configuration of the plant or its operation. The proposed change, based upon the results of the analysis, deletes the requirement to monitor toxic gas hazardous cargo and report the results to the NRC. Therefore, since the risk from toxic gas shipments along I-5 has been demonstrated to be and to remain acceptably low, operation of the facility in accordance with this proposed change will not involve a significant increase in the probability or consequences of accidents 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 plant or its operation. Therefore, operation of the facility in accordance with this 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

Based upon SCE's analysis, the risk to control room habitability from toxic gas shipments along I-5 has been demonstrated to be and to remain acceptably low for the duration of the Unit 3 operating license. Elimination of the requirement to monitor and periodically report this traffic is not needed to assure the habitability of the San Onofre Units 2 and 3 control room. Therefore, operation of the facility in accordance with this proposed change will not involve a significant 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 (vi) relates to a change which either may result in some increase in 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 refinement of a previously used calculation model or design method.

In this case, the pertinent acceptance criteria are found in SRP Section 2.2.3, "Evaluation of Potential Accidents." The acceptance criteria are based on meeting the relevant requirements of 10 CFR Part 100 as it relates to the factors to be considered in the evaluation of sites. These requirements indicate that reactors should reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. Specifically, the expected rate of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines of approximately 10^{-6} per year is acceptable if, when combined with reasonable qualatative arguments, the realistic probability can be shown to be lower.

An accident involving hazardous substances which could result in a toxic gas release and subsequent uninhabitability of the San Onofre Units 2 and 3 control room is considered to be an initiating event which could lead to potential consequences in excess of 10 CFR Part 100 exposure guidelines. With respect to the toxic gas release, a number of conservative assumptions were made relative to the toxic cloud's (resulting from an accident and a spill) ability to reach the control room intake. These assumptions are the same as those assumptions used in the analysis of individual gases considered previously and documented in the FSAR. Beyond uninhabitability of the control room, a number of other independent initiating events would be required to occur which would have to result in core damage, breach of the Reactor Coolant Pressure Boundary and failure of the containment structure in order to result in offsite exposure. SCE's analysis of the toxic gas hazard from shipments along I-5 demonstrates that the annual probability of the SONGS 1,2 & 3 control room becoming uninhabitable due to accidental release of <u>all</u> toxic chemicals is less than a medium value of 10^{-6} per year. This result well exceeds the SRP acceptance criteria in that the expected rate of occurrence of potential offsite exposure which might result during this condition is considered to be several orders of magnitude below the 10^{-6} value. In addition, the SCE analysis demonstrates that this probability of control room inhabitability remains acceptably low during the remainder of the Unit 3 operating license. As a result of this analysis, further highway

surveys to monitor toxic gas hazard traffic are unnecessary and may be discontinued.

The proposed change is similar to Example (vi) in that the proposed change may result in an insignificant increase in the probability of a previously analyzed accident, but where the results of the change are clearly within all acceptable criteria with respect to Standard Review Plan Section 2.2.3.

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; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; 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.

ATTACHMENT A

UNIT 2: EXISTING TECHNICAL SPECIFICATION

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the NRC Regional Administrator within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

HAZARDOUS CARGO TRAFFIC REPORT

6.9.1.14 Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

ATTACHMENT B

UNIT 2: PROPOSED TECHNICAL SPECIFICATION

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the NRC Regional Administrator within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

HAZARDOUS CARGO TRAFFIC REPORT

6.9.1.14 Explosive and flamability hazard cargo traffic on Interstate 5 (I-5) and hazardous cargo traffic on the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

ATTACHMENT C

UNIT 3: EXISTING TECHNICAL SPECIFICATION

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the NRC Regional Administrator within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

HAZARDOUS CARGO TRAFFIC REPORT

6.9.1.14 Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

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ATTACHMENT D

UNIT 3: PROPOSED TECHNICAL SPECIFICATION

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the NRC Regional Administrator within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

HAZARDOUS CARGO TRAFFIC REPORT

6.9.1.14 Explosive and flamability hazard cargo traffic on Interstate 5 (I-5) and hazardous cargo traffic on the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC Regional Administrator within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-237

This is a request to revise Technical Specification 3/4.2.7, "Axial Shape Index."

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 Specification (TS) 3/4.2.7, "Axial Shape Index" and its associated basis. Technical Specification 3.2.7 is provided to ensure that the actual value of the axial shape index (ASI) is maintained within the range assumed as an initial condition in the safety analyses. The range assumed in the safety analyses is $-0.3 \leq ASI \leq 0.3$. ASI is a measure of the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers. The ASI can be calculated utilizing either the Core Operating Limit Supervisory System (COLSS) or any operable Core Protection Calculator (CPC) channel. The real time monitoring capability and accuracy of COLSS allows COLSS to monitor power limit margins closely (using incore, self-powered, rhodium detectors).

The proposed change to this Specification is required to support Cycle 4 operation (24 month cycle versus the current 18 month cycles). Analysis of COLSS uncertainties has shown that the axial shape uncertainty increases from $\pm$ 0.02 to $\pm$ 0.03. This is due primarily to the effect of increased cycle length on the measurement uncertainties associated with the incore detectors.

Technical Specification 3.2.7 currently states that the ASI be maintained within the COLSS OPERABLE limits ($-0.28 \leq ASI \leq 0.28$) or the COLSS OUT OF SERVICE (CPC) limits ($-0.20 \leq ASI \leq 0.20$). The current COLSS OPERABLE limits for the ASI are based on the value assumed in the safety analyses ($-0.3 \leq ASI \leq 0.3$) taking into account the axial shape uncertainty ($\pm$ 0.02). For Cycle 4 operation, the axial shape uncertainty increases to $\pm$ 0.03. Therefore, the COLSS ASI alarm limit will be changed from $-0.28 \leq ASI \leq 0.28$ to $-0.27 \leq ASI \leq 0.27$. To reflect this, the proposed change would replace the numerical limit associated with the COLSS OPERABLE limit with a requirement to maintain the COLSS calculated ASI within the COLSS ASI alarm limits.

Surveillance Requirement 4.2.7 will also be revised. Currently, Section 4.2.7 only requires that ASI be determined to be within its limits once per 12 hours using COLSS or a CPC when COLSS is out of service. The proposed change will more clearly reflect that COLSS continuously monitors ASI when it is in

service. Specifically, ASI will be required to be continuously monitored and determined to be within its limit with COLSS, or with COLSS out of service, ASI will be required to be verified within its limit at least once every 12 hours using any operable CPC channel.

The bases for this Specification is expanded to include discussions of the 20% minimum power limitation in Mode 1, what ASI is and the two methods by which this parameter is calculated. The bases currently states that this Specification ensures that the actual value of the ASI is maintained within the range of values used in the safety analyses.

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 will qualify rather than quantify the limit established for the ASI when calculated by the COLSS. The ASI is an initial condition assumed in the safety analyses such that $-0.3 \leq ASI \leq 0.3$. When utilizing COLSS to determine ASI, the ASI is continuously calculated and compared to the parameter specified in the limiting condition for operation. If the value is exceeded, COLSS alarms are initiated. The alarm setpoints take into account COLSS uncertainties. Thus, the safety analyses assumptions are unaffected by the proposed change. Although this parameter would no longer be specified in the LCO, it will still remain in effect. The ASI safety setting is selected so that: 1) no safety limit will be exceeded as a result of an anticipated operational occurrence and 2) the consequences of a design basis accident will be acceptable. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any 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

   Operation of San Onofre in accordance with this proposed change will not create the possibility of a new or different kind of accident from any previously evaluated accident. ASI is specified as an initial condition in the Safety Analyses. This parameter will not be changed nor will it be exceeded. The LCO will be revised to

change the numerical limit to a requirement to maintain the COLSS calculated ASI within the COLSS ASI alarm limits. 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 change will revise the numerical limit associated with the ASI with a requirement to maintain ASI within the COLSS ASI alarm limit. The COLSS ASI alarm limit is set below the value assumed in the safety analyses accounting for COLSS uncertainties. The proposed change will ensure that this requirement is met. Therefore, the proposed change will not involve a significant reduction in 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 (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 are involved.

The proposed change, described above, is required for Cycle 4 operation. Analysis of COLSS uncertainties has shown that the axial shape uncertainty increases from $\pm$ 0.02 to $\pm$ 0.03. Rather than revise the numerical limit associated with the ASI, when calculated by COLSS, the proposed change will replace the limit with a requirement to maintain the COLSS calculated ASI within the COLSS ASI alarm limit. Because the change is required for a new cycle of operation, example (iii) applies as the proposed change relates to a nuclear core reloading.

The Surveillance Requirement associated with the LCO will also be revised. The core average ASI will be required to be monitored continuously by COLSS and determined to be within its limit. With COLSS out of service, the ASI will be verified within its limit at least once per 12 hours utilizing any operable CPC channel. The proposed change is similar to example (iii) because the change results from a nuclear core reloading.

<u>Safety and Significant Hazards Determination</u>

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 change; 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.

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ATTACHMENT A

(Existing Technical Specification)

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 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 B

(Proposed Technical Specification)

POWER DISTRIBUTION LIMITS

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. With COLSS operable, the COLSS ASI alarm limit.

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit by continuosly monitoring the ASI with COLSS, or with COLSS OUT OF SERVICE, by verifying at least once per 12 hours that the core average ASI is within the COLSS OUT OF SERVICE ASI limit using any operable CPC channel.

*See Special Test Exception 3.10.2.

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 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.

3/4.2.7 AXIAL SHAPE INDEX

The Axial Shape Index (ASI) is a measure of the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers. This specification is provided to ensure that the core average ASI is maintained within the range of values assumed as an initial condition in the safety analyses. This range is specified as $-0.3 \leq ASI \leq 0.3$.

The ASI can be determined by utilizing either the Core Operating Limit Supervisory System (COLSS) or any operable Core Protection Calculator (CPC) channel. The real time monitoring capability and accuracy of COLSS allows COLSS to monitor power limit margins closely. Consequently, the ASI limit is broader than it would be with the same core without COLSS. The COLSS continuously calculates the ASI and compares the calculated value to the parameter established for the COLSS ASI alarm limit. In addition, there is an uncertainty associated with the COLSS calculated ASI, therefore the COLSS ASI alarm limit includes this uncertainty. If the LCO is exceeded, COLSS alarms are initiated. The ASI safety setting is selected so that no safety limit will be exceeded as a result of an anticipated operational occurrence, and so that the consequence of a design basis accident will be acceptable.

AXIAL SHAPE INDEX (continued)

With COLSS out of service, any operable CPC channel may be used to calculate the ASI (using three axially spaced excore detectors). The axial shape synthesis in the CPC's shows the relative power produced as a function of core height in each third of the core. Due to the uncertainty associated with the CPC estimate, the ASI is restricted to a smaller range than the range calculated using the COLSS.

The 20% rated thermal power threshold is imposed due to the inaccuracy of the neutron flux detector below the threshold. Core noise level is too large to obtain usable detector readings.

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 Specification)

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq ASI \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI +0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

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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 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 D

(Proposed Technical Specification)

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq ASI \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

POWER DISTRIBUTION LIMITS

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. With COLSS operable, the COLSS ASI alarm limit.

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq ASI \leq +0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit by continuosly monitoring the ASI with COLSS, or with COLSS OUT OF SERVICE, by verifying at least once per 12 hours that the core average ASI is within the COLSS OUT OF SERVICE ASI limit using any operable CPC channel.

*See Special Test Exception 3.10.2.

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 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.

3/4.2.7 AXIAL SHAPE INDEX

The Axial Shape Index (ASI) is a measure of the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers. This specification is provided to ensure that the core average ASI is maintained within the range of values assumed as an initial condition in the safety analyses. This range is specified as $-0.3 \leq ASI \leq 0.3$.

The ASI can be determined by utilizing either the Core Operating Limit Supervisory System (COLSS) or any operable Core Protection Calculator (CPC) channel. The real time monitoring capability and accuracy of COLSS allows COLSS to monitor power limit margins closely. Consequently, the ASI limit is broader than it would be with the same core without COLSS. The COLSS continuously calculates the ASI and compares the calculated value to the parameter established for the COLSS ASI alarm limit. In addition, there is an uncertainty associated with the COLSS calculated ASI, therefore the COLSS ASI alarm limit includes this uncertainty. If the LCO is exceeded, COLSS alarms are initiated. The ASI safety setting is selected so that no safety limit will be exceeded as a result of an anticipated operational occurrence, and so that the consequence of a design basis accident will be acceptable.

AXIAL SHAPE INDEX (continued)

With COLSS out of service, any operable CPC channel may be used to calculate the ASI (using three axially spaced excore detectors). The axial shape synthesis in the CPC's shows the relative power produced as a function of core height in each third of the core. Due to the uncertainty associated with the CPC estimate, the ASI is restricted to a smaller range than the range calculated using the COLSS.

The 20% rated thermal power threshold is imposed due to the inaccuracy of the neutron flux detector below the threshold. Core noise level is too large to obtain usable detector readings.

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.