Attachment A

Unit 2

Existing Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I.   TYPE I ADDRESSABLE CONSTANTS

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

Attachment B

Unit 2

Proposed Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	< 0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.02
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted
104	PCALIB	Calorimetric Power	≤ 102.0

Attachment C

Unit 3

Existing Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

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Attachment D

Unit 3

Proposed Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

NOV 1 5 1982

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-150

This is a request to revise Technical Specification 3.2.7, "Power Distribution Limits - Axial Shape Index," for San Onofre Nuclear Generating Station Units 2 and 3.

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) 3.2.7, "Power Distribution Limits - Axial Shape Index," which specifies the Axial Shape Index (ASI) limit for power operation (Mode 1) with reactor power level greater than 20% RATED THERMAL POWER. ASI is a measure of power distribution within the reactor core and has a direct effect on thermal margin. The need for an ASI Limiting Condition for Operation (LCO) comes from the requirements that reactor design include appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

TS 3.2.7 establishes the ASI LCO. The proposed change revises the ASI LCO from its current ASI bounds of $-0.28 \le ASI \le +0.50$ to $-0.28 \le ASI \le +0.28$ with the Core Operating Limit Supervisory System (COLSS) in service and $-0.20 \le ASI \le +0.50$ to $-0.20 \le ASI \le +0.20$ with COLSS out of service. (Note: COLSS is a monitoring system used as an aid to the operator.) This proposed change will restrict the ASI band available to the operator.

The safety analysis performed in support of the Unit 2 Cycle 2 reload effort uses assumptions that are consistent with this proposed change to the ASI LCO. This analysis is presented in detail in the Reload Analysis Report for Cycle 2. The analysis results are clearly within all acceptance criteria. Further, it is pointed out that the proposed change constitutes an additonal limitation or restriction such that Cycle 2 safety analysis assumptions with respect to the ASI LCO are bounded by the assumptions used in the Cycle 1 safety analysis.

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 constitutes an additional limitation or
    restriction which was not previously in effect and, therefore, 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

    The proposed change does not involve any changes to operating
    procedures and, therefore, 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 constitutes an additional limitation or
    restriction which was not previously in effect and, therefore, 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 (11) relates to
a change that constitutes an additional limitation, restriction, or control
not presently included in the technical specifications: for example, a more
stringent surveillance requirement.

Since the change does constitute an additional limitation or restriction not
presently included in the Technical Specifications, the change will not result
in an increase in the probability or consequences of any accident previously
evaluated, nor will it result in a reduction in safety margin. Further, it
does not create the possibility of a new or different type of accident.
Therefore, the proposed change does not involve a 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 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.

PWS:2414F

Attachment A

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.   COLSS OPERABLE
        $-0.28 \leq ASI \leq + 0.50$

    a.   COLSS OUT OF SERVICE (CPC)
        $-0.20 \leq ASI \leq + 0.50$

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.

Attachment B

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

    a.   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.

Attachment C

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.   COLSS OPERABLE
            $-0.28 \leq ASI \leq +0.50$

    a.   COLSS OUT OF SERVICE (CPC)
            $-0.20 \leq ASI \leq +0.50$

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.

Attachment D

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.   COLSS OPERABLE
         $-0.28 \le ASI \le + 0.28$

   a.   COLSS OUT OF SERVICE (CPC)
         $-0.20 \le ASI \le + 0.20$

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.

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DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-151

This is a request to rev.ise Figure 3.1-2, "CEA Insertion Limits vs. Fraction
of Allowable Thermal Power," of Technical Specification 3.1.3.6, "Reactivity
Control Systems - Regulating CEA Insertion Limits," for San Onofre Nuclear
Generating Station Units 2 and 3.

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 Figure 3.1-2, "CEA Insertion Limits vs.
Fraction of Allowable Thermal Power," of Technical Specification (TS) 3.1.3.6,
"Reactivity Control Systems - Regulating CEA Insertion Limits," which
specifies the withdrawal sequence and Power Dependent Insertion Limits (PDIL)
for the regulating Control Element Assembly (CEA) groups. The need for a
specified withdrawal sequence and PDIL comes from the requirement that reactor
design include appropriate margin to assure that specified acceptable fuel
design limits are not exceeded during any condition of normal operation,
including anticipated operational occurrences. To this end, TS 3.1.3.6 helps
to 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.

TS 3.1.3.6 establishes the withdrawal sequence and PDIL. The proposed change
will revise the Short Term Steady State Insertion Limit and Transient
Insertion Limit specified by Figure 3.1-2. The Long Term Steady State
Insertion Limit will remain unchanged. The proposed change to the Short Term
Steady State Insertion Limit and Transient Insertion Limit constitutes an
additional limitation or restriction which is not included in the existing
Technical Specifications, but is included as an assumption in the Cycle 2
accident and transient analysis.

This restriction is imposed in order to reserve more margin for Steam System
Piping Failure Inside and Outside Containment (Standard Review Plan [SRP]
Section 15.1.5) and the Spectrum of Rod Ejection Accidents (SRP Section
15.4.8). The results of these two accidents for Cycle 2 are typically more
adverse that the results for Cycle 1 due to inherent differences between a
first cycle core and reload cores. The safety analysis performed in support
of the Cycle 2 reload effort, which includes the two accidents specifically
mentioned above, uses assumptions that are consistent with the proposed change

to the PDIL. The safety analysis results were clearly within all acceptance criteria. Further, it is pointed out that the proposed change constitutes an additional limitation or restriction not included in the existing technical specifications.

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 constitutes an additional limitation or restriction which was not previously in effect and, therefore, 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

    The proposed change does not involve any changes to operating procedures and, therefore, 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 constitutes an additional limitation or restriction which was not previously in effect and, therefore, 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 (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement.
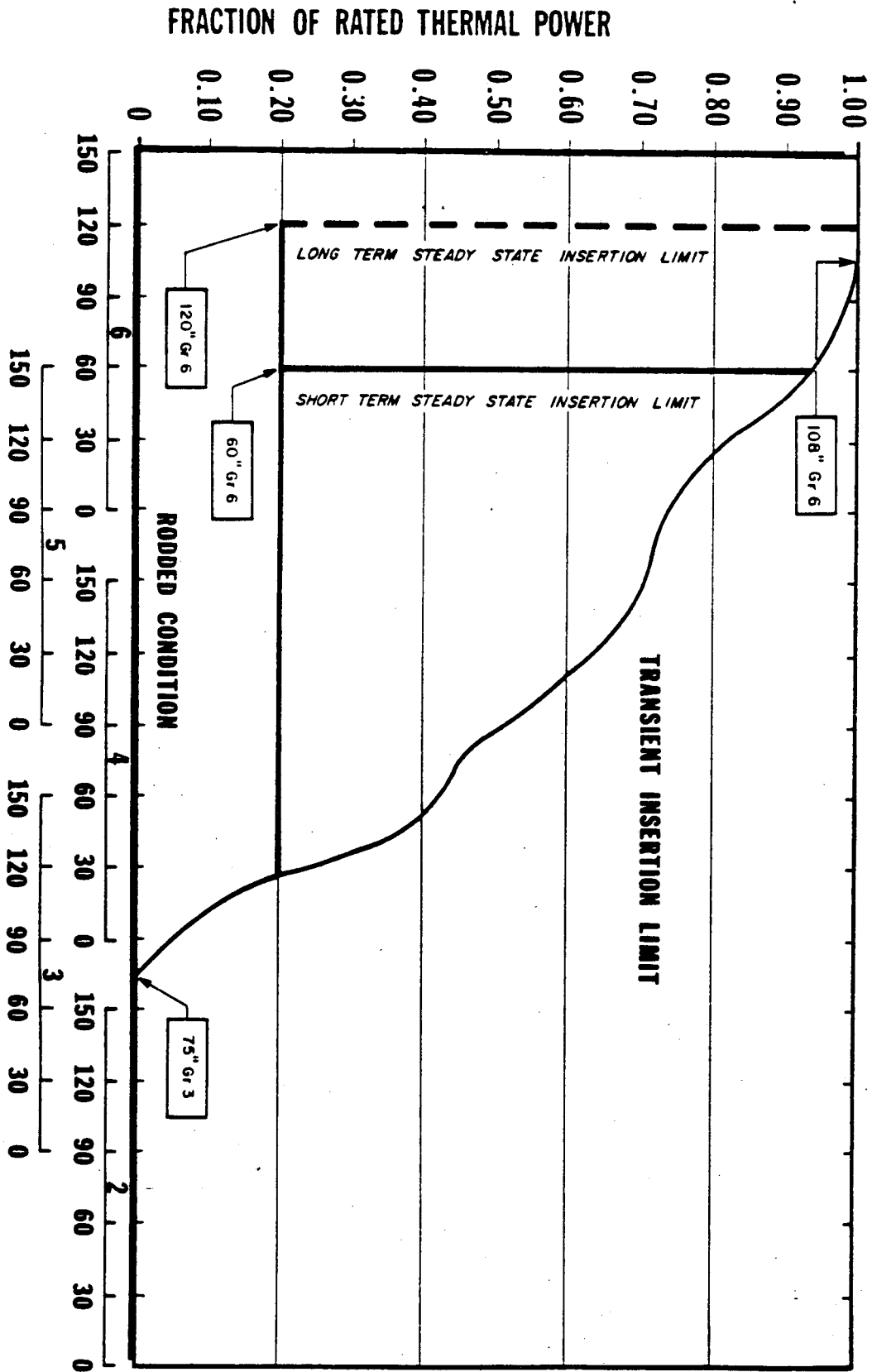
Since the proposed change does constitute an additional limitation or restriction not presently included in the technical specifications, it will not result in an increase in the probability or consequences of any accident previously evaluated, nor will it result in a reduction in safety margin. Further, the proposed change will not create the possibility of a new or different kind of accident. Therefore, the proposed change does not involve a 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 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.

PWS:2406F

Attachment A

FEB 17 1982

FRACTION OF RATED THERMAL POWER

0.10   0.20   0.30   0.40   0.50   0.60   0.70   0.80   0.90   1.00

LONG TERM STEADY STATE INSERTION LIMIT

120" Gr 6

SHORT TERM STEADY STATE INSERTION LIMIT

60" Gr 6

108" Gr 6

RODDED CONDITION

TRANSIENT INSERTION LIMIT

75" Gr 3

CEA WITHDRAWAL - INCHES

Figure 3.1-2

Figure 3.1-2

Attachment B

FIGURE 3.1-2

SAN ONOFRE-UNIT 2   3/4   1-24
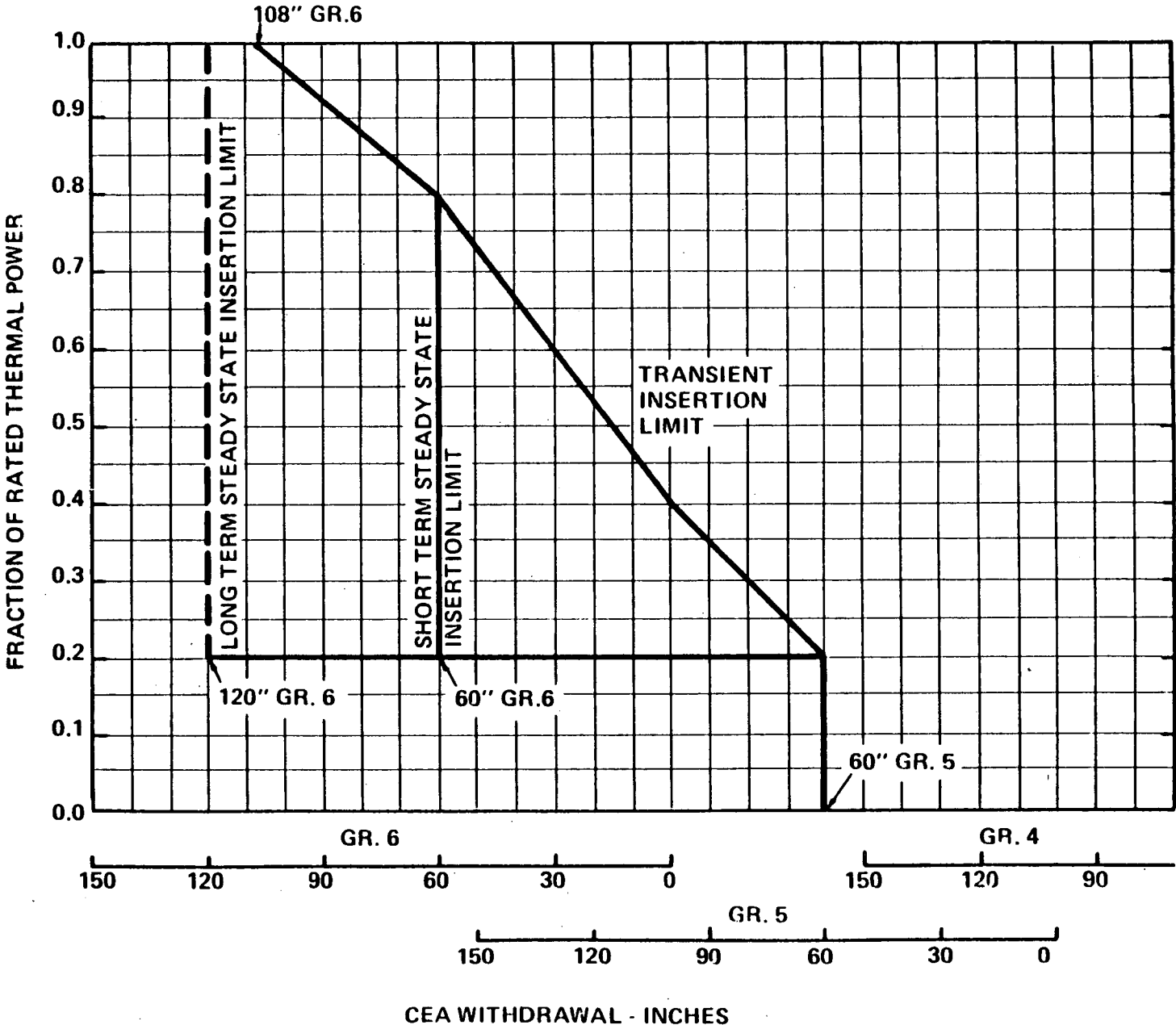
Attachment C

Figure 3.1-2
CEA insertion limits vs fraction
of allowable thermal power

Attachment D

FIGURE 3.1-2

This is a request to revise Technical Specification 2.2.2, Core Protection Calculator Addressable Constants (Table 2.2-2).

Description

The proposed change would revise Table 2.2-2 of Technical Specification 2.2.2, Core Protection Calculator (CPC) Addressable Constants. The CPC is an integral part of the reactor protection system. Some CPC addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level and radial peaking factors. Other CPC addressable constants allow inclusion of allowances for measurement uncertainties or inoperable equipment. Specifically, the proposed change adds the addressable constant point ID Number 103, Reactor Power Cutback Time Limit, to Table 2.2-2. The CPC algorithms which require the Reactor Power Cutback Time Limit is a small part of a larger CPC software package update provided to SONGS by Combustion Engineering (C-E), and represents a standard software package for C-E CPC's. The SONGS plant does not contain the hardware necessary for reactor power cutback, thus, the proposed addition of Point 103 does not have any effect on CPC function. The new addressable constant will be set to zero in the data base.

Some C-E reactors include a Reactor Power Cutback (RPC) System designed to eliminate the power imbalance without a trip after a loss of load. On SONGS 2, CPC modifications have been made to more accurately handle such transients without an RPC system, and also to avoid an unneeded trip. Even though SONGS does not have an RPC system, the RPC algorithms were included in the SONGS CPC and CEAC update in order to reduce the differences with other installed CPC/CEAC systems. The effect of those algorithms will be nullified through setting the data base and addressable constants associated with the RPC algorithm to zero. For more detailed information, see References 1 and 2.

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

Safety Analysis

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

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

   Response: No

   Since SONGS 2&3 do not have an RPC, the constants (addressable and non-addressable) in the RPC algorithm of the CPC will be set to zero. Therefore, the proposed change does not, in any way, affect the operation of the facility. Hence, there is no increase in the probability or consequences of an accident previously evaluated. In addition, should the value of this addressable constant inadvertently be changed from zero, the protection functions of the SONGS CPC system will continue unperturbed.

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

   Response: No

   No change to operating procedures is involved, thus, no new path is created which may lead to a new or different kind of accident.

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

   Response: No

   Since the constants (addressable and non-addressable) in the RPC algorithm of the CPC will be set to zero and the algorithm will therefore not function for SONGS 2&3, the proposed change does not, in any way, affect the operation of the facility.

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) describes a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method. The proposed change is similar to example (vi) of 48 FR 14870 in that it provides for future refinement of the CPC by the addition of algorithms to support a Reactor Power Cutback System. At present, the necessary hardware for an RPC system is not installed at SONGS 2 and the algorithms are deactivated by use of appropriate addressable and non-addressable constants. The approved CPC Software change procedures (see References 1 and 2) are used to verify that the RPC algorithms have no effect on CPC performance and plant safety margins.

Safety and Significant Hazards Determination

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

References

1. CEN-39(A)-P, Revision 02, "The CPC Protection Algorithm Software Change Procedure," December 2, 1978.

2. CEN-39(A)-P, Supplement 1-P, Revision 01, January 1979.

GvN:2411F

Attachment A

Unit 2

Existing Technical Specification

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

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

Attachment B

Unit 2

Proposed Technical Specification

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

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

Attachment C

Unit 3

Existing Technical Specification

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. <u>TYPE II ADDRESSABLE CONSTANTS (Continued)</u>

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

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Attachment D

Unit 3

Proposed Technical Specification

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

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

NOV 1 5 1982

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-153

This is a request to revise Technical Specification 3.1.1.3, "Reactivity Control Systems - Moderator Temperature Coefficient," for San Onofre Nuclear Generating Station Units 2 and 3.

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) 3.1.1.3, "Reactivity Control Systems - Moderator Temperature Coefficient," which specifies the Moderator Temperature Coefficient (MTC) limits for Power Operation and Startup Modes (Modes 1 and 2, respectively). MTC is a measure of the effect that reactor coolant temperature has on reactivity, which in turn effects reactor power (i.e., in the presence of a negative MTC, a decrease in temperature will cause an increase in power). The need for an MTC Limiting Condition for Operation (LCO) comes from the requirement that reactor design include appropriate limits on the potential amount and rate of reactivity increase.

TS 3.1.1.3 establishes the MTC LCO. The proposed change will revise the positive MTC limit from $<0.13 \times 10^{-4}$ delta $k/k/^\circ F$ for all reactor power levels to <0.0 delta $k/k/^\circ F$ for reactor power levels $>70\%$ RATED THERMAL POWER and $<0.5 \times 10^{-4}$ delta $k/k/^\circ F$ for reactor power levels $\leq 70\%$ RATED THERMAL POWER. This revision is consistent with the assumptions used in the Cycle 2 accident and transient analysis. For reactor power levels $>70\%$ RATED THERMAL POWER, the proposed change constitutes an additional limitation or restriction not included in the existing technical specifications. For reactor power levels $\leq 70\%$ RATED THERMAL POWER, the proposed change will result in a slight broadening of the allowed MTC band.

The safety analysis performed in support of the Cycle 2 reload effort uses assumptions with respect to the MTC that are consistent with the proposed change to the MTC LCO. The safety analysis results, which include the effects of the proposed change, are clearly within all acceptance criteria. These results are presented in detail in the Reload Analysis Report for Cycle 2. For powers $>70\%$ RATED THERMAL POWER, the proposed change to the MTC LCO constitutes an additional limitation or restriction such that Cycle 2 safety analysis assumptions with respect to the MTC LCO are bounded by the

assumptions used in the Cycle 1 safety analysis. For powers $\leq 70\%$ RATED THERMAL POWER, the proposed change will allow low power operation near the beginning of cycle (BOC) for future cycles with high soluble boron concentrations. Further, it is pointed out that both Cycle 1 and Cycle 2 safety analyses assume an MTC of $+.5 \times 10^{-4}$ delta k/k/$^{\circ}$F for both BOC analyses, even though the Cycle 1 MTC LCO is more restrictive.

Safety Analysis

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

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

    Response: No

    For power $> 70\%$ RATED THERMAL POWER, the proposed change constitutes an additional limitation or restriction which was not previously in effect and, therefore, will not involve an increase in the probability or consequences of any accident previously evaluated. For power $\leq 70\%$ RATED THERMAL POWER, the proposed change is incorporated as an assumption into the Cycle 2 safety analysis. The events most affected by the proposed change are those characterized by an increase in primary temperature. The details concerning the analysis of these events are presented in the Reload Analysis Report for Cycle 2. This analysis demonstrates that although the proposed change may be perceived to slightly increase in some way the consequences of an accident, the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan (SRP). 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

    The proposed change does not involve any changes to operating procedures and, therefore, 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

    For power $> 70\%$ RATED THERMAL POWER, the proposed change constitutes an additional limitation or restriction which was not previously in

effect and, therefore, will not involve a significant reduction in a margin of safety. For power ≤70% RATED THERMAL POWER, the proposed change is incorporated as an assumption into the Cycle 2 safety analysis. The events most affected by the proposed change are those characterized by an increase in primary temperature. The details concerning the analysis of these events are presented in the Reload Analysis Report for Cycle 2. This analysis demonstrates that although the proposed change may be perceived to slightly reduce a margin of safety, the results of the proposed change are clearly within all acceptable criteria with respect to the system or component specified in the SRP.

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 (ii) relates to a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example, a more stringent surveillance requirement. Example (vi) relates to a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

The part of the proposed change which is applicable during operation at powers >70% RATED THERMAL POWER is similar to Example (ii), because it constitutes an additional limitation, restriction, or control not presently included in the technical specifications. The part of the proposed change which is applicable during operation at powers ≤70% RATED THERMAL POWER is similar to Example (vi). The Anticipated Operational Occurrences (AOO) and Postulated Accidents that are more adverse in the presence of a positive MTC are affected by this part of the proposed change. Those AOO's and Postulated Accidents specified above which also tend to be limiting at zero or low powers include the Rod Ejection Accident (Standard Review Plan [SRP] Section 15.4.8), the Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition (SRP Section 15.4.1), the Loss of Non-Emergency AC Power to the Station Auxiliaries (including a 4-pump Loss of Flow) (SRP Section 15.2.6), and the Reactor Coolant Pump Seizure and Reactor Coolant Pump Shaft Break (SRP Section 15.3.4). During AOO's the acceptance criteria can be summarized as requiring that no fuel design limits be violated. During Postulated Accidents the acceptance criteria generally is that the core be maintained in a coolable geometry and that doses at the site boundary be within specified limits. The results of all AOO and Postulated Accident analyses for Cycle 2, including those specifically mentioned above, are clearly within all acceptance criteria with respect to the system or component specified in the SRP. Based on the discussion provided above, the proposed change is considered not to involve a 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 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.

PWS:2404F

Attachment A

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.13×10^{-4} delta k/k/°F, and

b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

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

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

#See Special Test Exception 3.10.2.

Attachment B

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

        Less positive than 0.0 delta k/k/^{o}F whenever THERMAL POWER is >70% of RATED THERMAL POWER, and

    b. **Less negative than -2.5 x 10^{-4} delta k/k/°F at RATED THERMAL POWER.**

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

    b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

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

---

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

#See Special Test Exception 3.10.2.

Attachment C

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

    a.   Less positive than 0.13×10^{-4} delta k/k/°F, and

    b.   Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

    b.   At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

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

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

#See Special Test Exception 3.10.2.

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Attachment D

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

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

    b. Less negative than -2.5×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

    b. At any THERMAL POWER, within 7 EFPD of reaching 40 EFPD core burnup.

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

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

#See Special Test Exception 3.10.2.

DESCRIPTION AND SAFETY ANALYSIS
OF PROPOSED CHANGE NPF-10/15-160

This is a request to revise Technical Specifications 2.1.1.1, "Safety Limits - Reactor Core - DNBR," and 2.2.1, "Limiting Safety System Settings - Reactor Trip Setpoints," and Technical Specification Bases 2.2.1, "Reactor Trip Setpoints," and 3/4.4.1, "Reactor Coolant Loops and Coolant Circulation," for San Onofre Nuclear Generating Station Units 2 and 3.

Existing Specifications:

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

Proposed Specifications:

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

Description

The proposed change revises Technical Specifications (TS) 2.1.1.1, "Safety Limits - Reactor Core - DNBR," and 2.2.1, "Limiting Safety System Settings - Reactor Trip Setpoints," and TS Bases 2.2.1, "Reactor Trip Setpoints," and 3/4.4.1, "Reactor Coolant Loops and Coolant Circulation," which specify the Departure from Nucleate Boiling Ratio (DNBR) Limiting Safety System Settings (LSSS). DNBR is a unitless value calculated for reactor core thermal-hydraulic conditions on a real-time basis from a Nuclear Regulatory Commission (NRC) approved empirical correlation. It is a measure of thermal margin. Maintaining core conditions such that DNBR is above a prescribed value helps to ensure that the fuel cladding will not overheat during Anticipated Operational Occurrences (AOO). The technical specifications affected by this change fall into two categories: first, the technical specifications establishing the Reactor Core Safety Limit for DNBR and the Reactor Protective Instrumentation Trip Setpoint Limit (or LSSS) which ensures that the established Safety Limit is not violated; and second, the various technical specification bases which quote the DNBR Safety Limit or LSSS.

TS 2.1.1.1 and Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," of TS 2.2.1 establish the DNBR Safety Limit and LSSS, respectively. The proposed change will revise both values from 1.20 to 1.31. The revision is brought about due to a change in the manner in which uncertainties are accounted for in the Departure from Nucleate Boiling (DNB) limit calculation. This revision will be implemented by a revised Core Protection Calculation (CPC) DNBR constant, changes to the CPC thermal margin algorithm constants, and the use of a consistent set of constants for the thermal-hydraulic computer code used in transient analysis. (Note: CPC's are an integral part of the reactor protective system. During AOO's, they provide a trip signal in time to prevent fuel damage.) The revision also deletes a portion of Note 5 of Table 2.2-1 which allows the lowering of the DNBR LSSS by an additional

0.01 to 1.19, because this flexibility is no longer needed. The changes to TS Bases 2.2.1 and 3/4.4.1 are for consistency only, since these TS Bases quote the DNBR Safety Limit or LSSS. These quotes are changed from 1.20 to 1.31 (or 1.19 to 1.31).

The requirement for a DNBR Safety Limit and a DNBR LSSS originates from 10 CFR 50 Appendix A, General Design Criterion (GDC) 10, "Reactor Design," which requires that reactor design include appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOO's. Specific criteria which must be met in order to meet the requirements of GDC 10 are described in Standard Review Plan (SRP) Section 4.4, "Thermal and Hydraulic Design." SRP Section 4.4 provides the following acceptable approach to meeting DNB criteria which follow from GDC 10:

> "For departure from nucleate boiling ratio (DNBR), critical heat flux ratio (CHFR), or critical power ratio (CPR) correlations there should be a 95% probability at the 95% confidence level that the hot rod in the core does not experience a departure from nucleate boiling or boiling transition condition during normal operation or anticipated operational occurrences."

The Cycle 1 CPC accommodates uncertainites in a combination of deterministic and statistical methods. System parameter uncertainties are deterministically built into the constants within the CPC Thermal Margin algorithm. Flow, temperature, pressure, and power measurement uncertainties were also treated deterministically. Cycle 2 implements a program for statistical combination of systems and state parameter uncertainties. This program includes the combination of system parameter uncertainties in a single adjustment of the DNBR limit and the combination of measurement uncertainties with CPC modeling errors in the calculation of certain CPC addressable constants.

The DNBR limit is increased for Cycle 2 to accommodate system parameter uncertainties at a 95/95 probability/confidence level. This results in a revised CPC DNBR constant, changes to the CPC Thermal Margin algorithm constants, and the use of a consistent set of constants for the thermal-hydraulic computer code used for transient analysis.

Use of the statistical method for the calculation of DNB limiting safety system setpoints will help to ensure that GDC 10 is met by providing a 95% probability at the 95% confidence level that the hot rod in the core will not experience a DNB or boiling transition condition during normal operation or AOO's. This method has received generic approval by NRC acceptance of the Combustion Engineering Standard Safety Analysis Report (CESSAR) for Palo Verde in "SER Related to Final Design Approval of CE Standard NSSS (CESSAR)," NUREG-0852, Supplement 2, September, 1983 (pp. 4-11 to 24). The statistical methods to be used for Cycle 2 are identical to those methods reviewed by the NRC for Palo Verde Nuclear Generating Station Unit 2 Cycle 1.

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 DNBR limits generated by both the Cycle 1 methodology and the statistical combination of uncertainties methodology are designed to provide a 95% probability at the 95% confidence level that the hot rod in the core will not experience a DNB or boiling transition condition during normal operation or AOO's. Since both methods meet the same criteria, 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

   The proposed change does not involve any changes to operating procedures or CPC algorithms and, therefore, 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 use of the statistical combination of uncertainties method provides a reduction in analytical conservatism of uncertainties only. The criterion for appropriate margin described in GDC 10 and the specific 95/95 probability/confidence DNBR criterion from SRP Section 4.4 remain unchanged. 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 (vi) relates to a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a

safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

The proposed change is similar to the above example for the following reasons. First, the proposed change will have little or no adverse effect on safety margin. For most events it is possible to demonstrate that the plant will possess greater safety margin for identical accidents or transients than can be demonstrated using Cycle 1 methodology. Specifically, the DNBR Safety Limit will be adjusted to meet the 95/95 probability/confidence requirements of SRP Section 4.4. With respect to the probability or consequences of a previously analyzed accident, the same beneficial effects apply. Second, the results of all safety analyses performed in support of Cycle 2 and specifically those which use DNBR as acceptance criteria are clearly within the limits specified in the SRP.

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.

PWS:2397F

Attachment A

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.20.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.20, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

   Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

   Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 19.95 kw/ft	$\leq$ 19.95 kw/ft
10. DNBR - Low	$\geq$ 1.20 (5)	$\geq$ 1.20 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.3%/sec (6)(8)	$\leq$ 0.315%/sec (6)(8)
b) Floor	$\geq$ 60% (6)(8)	$\geq$ 55% (6)(8)
c) Step	$\leq$ 10% (6)(8)	$\leq$ 13% (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

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 1.20. A DNBR trip setpoint of 1.19 is allowed provided that the difference is compensated by an increase in the addressable constants BERR1 for CPC and EPOL2 for COLSS.

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

     FLOOR is the minimum value of the trip setpoint.

     SIFP 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.

BASES

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SETPOINTS

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

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.20 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

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

Manual Reactor Trip

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

BASES

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

    a.   Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

    b.   Reactor Coolant System pressure from pressurizer pressure measurement;

    c.   Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

    d.   Radial peaking factors from the position measurement for the CEAs;

    e.   Reactor coolant mass flow rate from reactor coolant pump speed;

    f.   Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.20 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit.
CPC uncertainties related to DNBR cover CPC input measurement uncertainties,
algorithm modelling uncertainties, and computer equipment processing uncer-
tainties. Dynamic compensation is provided in the CPC calculations for the
effects of coolant transport delays, core heat flux delays (relative to changes
in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated
below and operation outside of these limits will result in a CPC initiated trip.

a. RCS Cold Leg Temperature-Low $\geq$ 495°F
b. RCS Cold Leg Temperature-High $\leq$ 580°F
c. Axial Shape Index-Positive $<$ +0.5
d. Axial Shape Index-Negative $>$ -0.5
e. Pressurizer Pressure-Low $\geq$ 1825 psia
f. Pressurizer Pressure-High $\leq$ 2375 psia
g. Integrated Radial Peaking Factor-Low $>$ 1.28
h. Integrated Radial Peaking Factor-High $\leq$ 4.28
i. Quality Margin-Low $<$ 0

The DNBR Trip setpoint in CPC and COLSS is 1.19. The values of the penalty
factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements
for tripping at other values of DNBR. The following formula is used to adjust
the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old} \left[1 + \Delta DNBR(\%) * \left|\frac{d(\% \ POL)}{d(\% \ DNBR)}\right| * 0.01\right]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$\left|d(\% \ POL)/d(\% \ DNBR)\right|$ = The absolute value of the most adverse derivative
of percent POL with respect to percent DNBR as
reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = \left(1 + \Delta DNBR(\%) * \left|\frac{d(\% \ POL)}{d(\% \ DNBR)}\right| * 0.01\right) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.20 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

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

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 235°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 235°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

Attachment B

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.31

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.31, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 19.95 kw/ft	$\leq$ 19.95 kw/ft
10. DNBR - Low	$\geq$ 1.31 (5)	$\geq$ 1.31 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.3%/sec (6)(8)	$\leq$ 0.315%/sec (6)(8)
b) Floor	$\geq$ 60% (6)(8)	$\geq$ 55% (6)(8)
c) Step	$\leq$ 10% (6)(8)	$\leq$ 13% (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

(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.

SIFP 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SETPOINTS

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

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

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

Manual Reactor Trip

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

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

    a.   Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

    b.   Reactor Coolant System pressure from pressurizer pressure measurement;

    c.   Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

    d.   Radial peaking factors from the position measurement for the CEAs;

    e.   Reactor coolant mass flow rate from reactor coolant pump speed;

    f.   Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.31 such that the decrease in actual core

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$> 495°F$
b.	RCS Cold Leg Temperature-High	$< 580°F$
c.	Axial Shape Index-Positive	$< +0.5$
d.	Axial Shape Index-Negative	> -0.5
e.	Pressurizer Pressure-Low	≥ 1825 psia
f.	Pressurizer Pressure-High	≤ 2375 psia
g.	Integrated Radial Peaking Factor-Low	> 1.28
h.	Integrated Radial Peaking Factor-High	< 4.28
i.	Quality Margin-Low	< 0

The DNBR Trip setpoint in CPC and COLSS is 1.31. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old} \left[1 + \Delta DNBR(\%) * \left|\frac{d(\% POL)}{d(\% DNBR)}\right| * 0.01\right]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$\left|d(\% POL)/d(\% DNBR)\right|$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = \left(1 + \Delta DNBR(\%) * \left|\frac{d(\% POL)}{d(\% DNBR)}\right| * 0.01\right) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

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

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^{6} lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 235°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 235°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

Attachment C

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.20.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.20, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

   Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

   Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High -		
Four Reactor Coolant Pumps Operating	$\leq$ 110.0% of RATED THERMAL POWER	$\leq$ 111.3% of RATED THERMAL POWER
3. Logarithmic Power Level - High (1)	$\leq$ 0.89% of RATED THERMAL POWER	$\leq$ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	$\leq$ 2382 psia	$\leq$ 2389 psia
5. Pressurizer Pressure - Low (2)	$\geq$ 1806 psia	$\geq$ 1763 psia
6. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
7. Steam Generator Pressure - Low (3)	$\geq$ 729 psia	$\geq$ 711 psia
8. Steam Generator Level - Low	$\geq$ 25% (4)	$\geq$ 24.23% (4)
9. Local Power Density - High (5)	$\leq$ 19.95 kw/ft	$\leq$ 19.95 kw/ft
10. DNBR - Low	$\geq$ 1.20 (5)	$\geq$ 1.20 (5)
11. Reactor Coolant Flow - Low		
a) DN Rate	$\leq$ 0.22 psid/sec (6)(8)	$\leq$ 0.231 psid/sec (6)(8)
b) Floor	$\geq$ 13.2 psid (6)(8)	$\geq$ 12.1 psid (6)(8)
c) Step	$\leq$ 6.82 psid (6)(8)	$\leq$ 7.231 psid (6)(8)
12. Steam Generator Level - High	$\leq$ 90% (4)	$\leq$ 90.74% (4)
13. Seismic - High	$\leq$ 0.48/0.60 (7)	$\leq$ 0.48/0.60 (7)
14. Loss of Load	Turbine stop valve closed	Turbine stop valve closed

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 1.20. A DNBR trip setpoint of 1.19 is allowed provided that the difference is compensated by an increase in the addressable constants BERRI for CPC and EPOL2 for COLSS.

(6) ON 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 ON 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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SETPOINTS

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

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of 1.20 and 19.95 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Seismic-High trip is generated by an open contact signal from a force balance contact device which is likewise not subject to analog type drifts. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

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

Manual Reactor Trip

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

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Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.20 such that the decrease in actual core

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DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$\geq$ 495°F
b.	RCS Cold Leg Temperature-High	$\leq$ 580°F
c.	Axial Shape Index-Positive	< +0.5
d.	Axial Shape Index-Negative	> -0.5
e.	Pressurizer Pressure-Low	$\geq$ 1825 psia
f.	Pressurizer Pressure-High	$\leq$ 2375 psia
g.	Integrated Radial Peaking Factor-Low	> 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	< 0

The DNBR Trip setpoint in CPC and COLSS is 1.19. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old} \left[1 + \Delta DNBR(\%) * \left| \frac{d(\% \; POL)}{d \; (\% \; DNBR)} \right| * 0.01 \right]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$d(\% \; POL)/d(\% \; DNBR)$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = \left(1 + \Delta DNBR(\%) * \left| \frac{d(\% \; POL)}{d \; (\% \; DNBR)} \right| * 0.01 \right) * (1 + EPOL2_{old}) - 1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps (RCPs) in operation, and maintain DNBR greater than 1.20 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than four reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

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

The restrictions on starting a reactor coolant pump in MODES 4 and 5 with one or more RCS cold legs less than or equal to 285°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 285°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 285°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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Attachment D

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.31.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.31, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

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TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

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

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.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR TRIP SETPOINTS

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

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

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

Manual Reactor Trip

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

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BASES

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a.   Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b.   Reactor Coolant System pressure from pressurizer pressure measurement;

c.   Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d.   Radial peaking factors from the position measurement for the CEAs;

e.   Reactor coolant mass flow rate from reactor coolant pump speed;

f.   Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than 1.31 such that the decrease in actual core

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DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	$> 495°F$
b.	RCS Cold Leg Temperature-High	$\leq 580°F$
c.	Axial Shape Index-Positive	$< +0.5$
d.	Axial Shape Index-Negative	> -0.5
e.	Pressurizer Pressure-Low	≥ 1825 psia
f.	Pressurizer Pressure-High	≤ 2375 psia
g.	Integrated Radial Peaking Factor-Low	> 1.28
h.	Integrated Radial Peaking Factor-High	≤ 4.28
i.	Quality Margin-Low	< 0

The DNBR Trip setpoint in CPC and COLSS is 1.31. The values of the penalty factors BERR1 (CPC) and EPOL2 (COLSS) may be adjusted to implement requirements for tripping at other values of DNBR. The following formula is used to adjust the CPC addressable constant BERR1:

$$BERR1_{new} = BERR1_{old} \left[1 + \Delta DNBR(\%)* \left|\frac{d(\% \ POL)}{d \ (\% \ DNBR)}\right| *0.01\right]$$

where:

$BERR1_{new}$ = new required value of BERR1,

$BERR1_{old}$ = present implemented value of BERR1,

$\Delta DNBR(\%)$ = percent increase in DNBR trip setpoint requirement,

$d(\% \ POL)/d(\% \ DNBR)$ = The absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

Similarly, for the COLSS addressable constant EPOL2:

$$EPOL2_{new} = \left(1 + \Delta DNBR(\%)*\left|\frac{d(\% \ POL)}{d \ (\% \ DNBR)}\right|*0.01\right)*(1 + EPOL2_{old})-1.0$$

where:

$EPOL2_{new}$ = new required value of EPOL2,

$EPOL2_{old}$ = present implemented value of EPOL2,

and the other terms are as previously defined.

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3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps (RCPs)in operation, and maintain DNBR greater than 1.31 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than four reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

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

The restrictions on starting a reactor coolant pump in MODES 4 and 5 with one or more RCS cold legs less than or equal to 285°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 285°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 285°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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This is a request to revise Technical Specification 2.2.2, Core Protection
Calculator Addressable Constants (Table 2.2-2).

Description

The proposed change would revise Table 2.2-2 of Technical Specification 2.2.2,
Core Protection Calculator (CPC) Addressable Constants. The CPC is an
integral part of the reactor protection system. Some CPC addressable
constants are provided to allow calibration of the CPC system to more accurate
indications of power level and radial peaking factors. Other CPC addressable
constants allow inclusion of allowances for measurement uncertainties or
inoperable equipment. Specifically, the proposed change redefines the CPC
addressable constant point ID Number 98. The addressable constant point ID 98
is currently defined as the "Temperature Shadowing Factor Correction
Multiplier". (Note: By "Temperature Shadowing" we mean the decalibration of
ex-core neutron flux power resulting from the changes in inlet coolant
density.) A modificaton to the CPC Temperature Shadowing Factor (TSF)
algorithm for Cycle 2 has resulted in the Temperature Shadowing Correction
Multiplier becoming fixed in the CPC software. The proposed change would
redefine the addressable constant point ID 98 as the "Reference Cold Leg
Temperature," consistent with the CPC TSF algorithm modifications and would
reclassify it as a Type I Addressable Constant (Type I implies requiring
periodic calibration). The proposed change combined with TSF modifications
would improve the thermal margin at nominal inlet temperature. At conditions
other than nominal conditions, the proposed change provides a more
conservative TSF.

The proposed change would revise the definition of CPC addressable constant
point ID number 98 from "Temperature Shadowing Correction Factor Multiplier"
to "Reference Cold Leg Temperature." This change is made in order to be
consistent with Temperature Shadowing Factor (TSF) algorithm improvements (see
Reference 1). TSF is used to correct the CPC neutron flux power for excore
detector decalibration effects resulting from changes in density of the
coolant passing between the reactor core and the neutron detectors. The TSF
algorithm was modified to include uncertainties directly in the calculations.
This improvement provides a conservative correction for temperature at
moderator temperatures above or below the inlet moderator temperature at which
the neutron flux power was last calibrated while providing a more accurate
indication of power near the calibration temperature. To accomplish this
correction, the coolant temperature at the time of the latest excore detector
calibration must be input as an addressable constant. Because of the
calibration requirements, CPC addressable constant point ID Number 98 is also
reclassified as a Type I Addressable Constant. The previous addressable
constant associated with this point ID no longer needs to be addressable and
is incorporated directly into the software. For more detailed information,
see Reference 1.

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

Safety Analysis

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

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

    Response: No

    The proposed change is designed to enhance the accuracy the CPC neutron flux power calculation by a more accurate treatment of uncertainties, thus avoiding any increase in the probability or consequences of an accident previously evaluated.

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

    Response: No

    No change to operating procedures is involved, thus no new path is created which may lead to a new or different kind of accident.

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

    Response: No

    The proposed amendment does not significantly alter the use of the CPC system to protect against operation of the reactor in a manner which would result in violation of the Specified Acceptable Fuel Design Limits. The change involves only a more detailed model of core power level measurement uncertainties. Thus, the proposed amendment maintains the same margin of safety during Cycle 2 operation as in Cycle 1.

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) describes a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method. The proposed change is similar to example (vi) of 48 FR 14870 in that the proposed change is a refinement of the previously used calculational model for correcting incore detector signals for the effects of temperature shadowing. Further, Cycle 2 Safety Analyses included the proposed change into the simulated CPC response to the Anticipated Operational Occurrences (AOO's) and Postulated Accidents which depend on the CPC to show protection. All Cycle 2 AOO's and postulated accidents were clearly within all acceptable criteria with respect to the system or component specified in the applicable Standard Review Plan. Furthermore, the proposed change enhances the reactor protection system's ability to meet the criteria specified in Standard Review Plan 7.2 "Reactor Trip System" in that it enhances the CPC's ability to sense accident conditions and to initiate the operation of systems and components important to safety.

Safety and Significant Hazards Determination

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

Reference

1. CEN-281(S)-P, "CPC/CEAC Software Modifications for San Onofre Nuclear Generating Station Units No. 2 and 3," June 1984.

GvN:2409F

Attachment A

Unit 2

Existing Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

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

Attachment B

Unit 2

Proposed Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I.   TYPE II ADDRESSABLE CONSTANTS (Continued)

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

Attachment C

Unit 3

Existing Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

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

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TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

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TABLE 2.2-2 (Continued)

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE II ADDRESSABLE CONSTANTS (Continued)

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

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DESCRIPTION OF PROPOSED CHANGE NPF-10/15-164
AND SAFETY ANALYSIS

This is a request to revise Technical Specification 2.2.2, Core Protection Calculator Addressable Constants (Table 2.2-2).

Description

The proposed change would revise Table 2.2-2 of Technical Specification 2.2.2, Core Protection Calculator (CPC) Addressable Constants. The CPC is an integral part of the reactor protection system. Some CPC addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level and radial peaking factors. Other CPC addressable constants allow inclusion of allowances for measurement uncertainties or inoperable equipment. Specifically, the proposed change revises the allowable value for the addressable constant point ID number 63 on Table 2.2-2, azimuthal tilt allowance (TR) (note: azimuthal power tilt is the power asymmetry between the azimuthally symmetric fuel assemblies). This change is made in order to be consistent with the Core Operating Limit Supervisory System (COLSS) azimuthal tilt algorithm modifications. (Note: COLSS provides reliable and continual information on the status of the reactor as an aid to the operator.) The proposed change would revise the minimum allowed value of addressable constant TR from 1.02 to 1.00. The proposed change broadens the range of TR values which can be used as addressable constants.

Currently, COLSS uses an "arithmetic average" technique to compute a core average azimuthal tilt value. Using this method, signal noise impact is enhanced by accumulating the magnitude component but ignoring the directional components of the tilt from each tilt group. The "planar vector average" technique performs a vector sum of the individual tilt estimates at each axial plane to calculate an average tilt estimate for each plane. The planar tilt estimates are then arithmetically averaged to obtain a total core average tilt. By introducing a planar vector average technique, the noise effects are reduced by allowing possible cancellation of some of the random components of noise. Thus, when there is no azimuthal tilt in the core, COLSS will yield an appropriately low (vector) tilt estimate. The reactor average vector tilt calculation has been demonstrated to agree well with the arithmetric average calculation in the presence of a true azimuthal tilt. The purpose of the lower minimum allowed value of the CPC azimuthal tilt multiplier, TR, is to reflect the reduced COLSS tilt estimate in situations when there is no appreciable azimuthal tilt in the core.

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

Safety Analysis

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

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

    Response: No

    The proposed change does not, in any way, affect the operation of the facility. The CPC trip functions remain unchanged since only the allowed range of a CPC addressable multiplier is affected. Hence, there is no increase in the probability or consequences of an accident previously evaluated.

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

    Response: No

    No change to operating procedures is involved, thus, no new path is created which may lead to a new or different kind of accident.

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

    Response: No

    The proposed amendment only broadens the allowable range for the TR addressable constant. Technical Specification criterion requiring the monitoring of tilt and incorporation in the CPC of the TR addressable constant greater than or equal to the measured value remains intact.

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) describes a change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all

acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method. The proposed change is similar to example (vi) of 48 FR 14870 in that the proposed change relates to a refinement of the previously used calculational model which estimates the azimuthal tilt in the reactor. Further, the change enhances the reactor protection system's ability to meet criteria specified in Standard Review Plan 7.2 "Reactor Trip System" in that it enhances the CPC's ability to sense accident conditions and to initiate the operation of systems and components important to safety.

Safety and Significant Hazards Determination

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

GvN:2408F

Attachment A

Unit 2

Existing Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I.   TYPE I ADDRESSABLE CONSTANTS

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

Attachment B

Unit 2

Proposed Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	< 0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.00
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

Attachment C

Unit 3

Existing Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

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

Attachment D

Unit 3

Proposed Technical Specification

TABLE 2.2-2

CORE PROTECTION CALCULATOR ADDRESSABLE CONSTANTS

I. TYPE I ADDRESSABLE CONSTANTS

POINT ID NUMBER	PROGRAM LABEL	DESCRIPTION	ALLOWABLE VALUE
60	FC1	Core coolant mass flow rate calibration constant	≤ 1.15
61	FC2	Core coolant mass flow rate calibration constant	0.0
62	CEANOP	CEAC/RSPT inoperable flag	0, 1, 2 or 3
63	TR	Azimuthal tilt allowance	≥ 1.00
64	TPC	Thermal power calibration constant	≥ 0.90
65	KCAL	Neutron flux power calibration constant	≥ 0.85
66	DNBRPT	DNBR pretrip setpoint	Unrestricted
67	LPDPT	Local power density pretrip setpoint	Unrestricted

This is a request to revise Section 3/4.10.4 and Bases - Special Test
Exceptions - Center CEA Misalignment of the Technical Specifications for San
Onofre Nuclear Generating Station, Units 2 and 3.

Description

The proposed change revises Technical Specification 3/4.10.4 and Bases -
Special Test Exceptions - Center CEA Misalignment which permits CEA
misalignment during Physics Tests as required to determine the isothermal
temperature coefficient, moderator temperature coefficient, and power
coefficient (these coefficients are a measure of the effects of changes in
temperature and power on reactivity). The need for such tests comes from the
requirement that a test program be established to demonstrate that the reactor
plant can be operated in accordance with the design requirements important to
safety. The proposed change includes an exception to permit insertion of
Regulating Control Rod Group 6 beyond the Transient Insertion Limit and a
surveillance requirement to continuously monitor Departure from Nucleate
Boiling Ratio (DNBR) during testing.

Section 3.10.4 establishes the Special Test Exceptions for the performance of
physics tests to determine the isothermal temperature coefficient, moderator
temperature coefficient, and power coefficient. Section 3.10.4 suspends
Technical Specifications 3.1.3.1 and 3.1.3.6 (the CEA Position and Regulating
CEA Insertion Limit - Limiting Conditions for Operation, respectively) and
allows the center CEA (CEA #1) to be purposely misaligned during these physics
tests. The proposed revision would allow Regulating Group #6 to be inserted
beyond its Transient Insertion Limit during this testing. In addition, a
surveillance requirement to continuously determine DNBR margin is added. The
test procedures for Cycle 2 (which are virtually identical to those used in
Cycle 1) require Group 6 to be moved and may result in the Transient Insertion
Limit being exceeded due to the application of a more restrictive Power
Dependent Insertion Limit (PDIL) for Cycle 2 (Figure 3.1-2 of the Technical
Specifications). A request for the more restrictive PDIL is before the NRC
for consideration and approval (Proposed Change NPF-10/15-151), forwarded by
separate correspondence. Another consideration is that future cycles at EOC
may require greater rod motion during testing due to the anticipated presence
of a more negative MTC than in Cycle 1.

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

Safety Analysis

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

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

    Response: No

    During the conduct of physics tests, very stringent surveillance requirements are in place; explicitly stated and enforced by the use of detailed operating procedures. This Technical Specification change does not affect the procedural limits or precautions, and allows only Group 6 movement beyond the Transient Insertion Limit under the controlled conditions established by the existing physics test procedures. These controlled conditions (continuous monitoring by incore detectors, etc.) ensure that the consequences of an accident will be limited to those reported in the Reload Analyses Report without the restrictions normally imposed by the PDIL. Therefore, the proposed change will 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

    No change to operating procedures is involved. Therefore, no new path is created which may lead to a new or different kind of accident.

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

    Response: No

    During the conduct of physics test, very stringent surveillance requirements are in place; explicitly stated and enforced by the use of detailed operating procedures. This Technical Specification change does not affect the procedural limits or precautions, and allows only Group 6 movement beyond the Transient Insertion Limit under the controlled conditions established by the existing physics test procedures. These controlled conditions (continuous monitoring by incore detectors, etc.) ensure that the margin of safety is not significantly reduced by the proposed change. The consequences of an accident during the physics tests will be limited to those reported in the Reload Analysis Report.

The commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48 FR 14870) of amendments that are considered least likely to involve significant hazards considerations. Example (vi) from the Federal Register discusses changes which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptance criteria with respect to the system or component specified in the Standard Review Plan (SRP); for example, a change resulting from the application of a small refinement of a previously used calculational model or design method. The accident which most depends on the PDIL to help ensure acceptable results at the power of concern during physics testing is the Spectrum of Rod Ejection Accident (SRP 15.4.8). At the reactor power range of concern, sufficient margin exists in the analysis of the CEA ejection accident that acceptable results can be demonstrated with Group 6 insertion beyond the Transient Insertion Limit (as required for the determination of reactivity coefficients).

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:2401F

Attachment A

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BASES

MAY 1 6 1983
AMENDMENT NO. 16

SPECIAL TEST EXCEPTIONS

3/4.10.4 CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

   a.   Only the center CEA (CEA #1) is misaligned, and

   b.   The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 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

4.10.4.1   The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2   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 Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 RADIATION MONITORING/SAMPLING

This special test exception permits fuel loading and reactor operation with radiation monitoring/sampling instrumentation calibration and quality assurance conforming to either FSAR procedures or Regulatory Guide 4.15 Rev 1, February 1979. This test exception is required to allow for a phased implementation of Regulatory Guide 4.15 Rev. 1, February 1979. Equivalent instrumentation, quality assurance and/or calibration is provided until full implementation of Regulatory Guide 4.15 Rev. 1, February 1979.

Attachment B

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<u>LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS</u>

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SPECIAL TEST EXCEPTIONS

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.4   The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be
suspended during the performance of PHYSICS TESTS to determine the isothermal
temperature coefficient, moderator temperature coefficient and power
coefficient provided:

   a.   Only the center CEA (CEA #1) is misaligned, or only Regulating CEA
        Group 6 is inserted beyond the Transient Insertion Limit of
        Specification 3.1.3.6; and

   b.   The limits of Specifications 3.2.1 and 3.2.4 are maintained and
        determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

   a.   Reduce THERMAL POWER sufficiently to satisfy the requirements of
        Specifications 3.2.1 and 3.2.4, or

   b.   Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1   The THERMAL POWER shall be determined at least once per hour during
PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or
3.1.3.6 are suspended and shall be verified to be within the test power
plateau.

4.10.4.2   The linear heat rate and DNBR Margin shall be determined to be
within the limits of Specifications 3.2.1 and 3.2.4, respectively, by
monitoring them continuously with the Incore Detector Monitoring System
pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS
above 5% of RATED THERMAL POWER in which the requirements of Specifications
3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEA's to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10/3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

This special test exception permits the center CEA to be misaligned or Regulating Group 6 inserted beyond the Transient Insertion Limit during PHYSICS TESTS required to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient.

3/4.10.5 RADIATION MONITORING/SAMPLING

This special test exception permits fuel loading and reactor operation with radiation monitoring/sampling instrumentation calibration and quality assurance conforming to either FSAR procedures or Regulatory Guide 4.15, Rev. 1, February 1979. This test exception is required to allow for a phased implementation of Regulatory Guide 4.15, Rev. 1, February 1979. Equivalent instrumentation, quality assurance and/or calibration is provided until full implementation of Regulatory Guide 4.15 Rev. 1, February 1979.

Attachment C

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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SPECIAL TEST EXCEPTIONS

3/4.10.4 CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

    a.   Only the center CEA (CEA #1) is misaligned, and

    b.   The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.6 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

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 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 Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

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3/4.10 SPECIAL TEST EXCEPTIONS

<u>BASES</u>

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

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<u>LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS</u>

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BASES

SPECIAL TEST EXCEPTIONS

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

    a.   Only the center CEA (CEA #1) is misaligned, or only Regulating CEA Group 6 is inserted beyond the Transient Insertion Limit of Specification 3.1.3.6; and

    b.   The limits of Specifications 3.2.1 and 3.2.4 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

    a.   Reduce THERMAL POWER sufficiently to satisfy the requirements of Specifications 3.2.1 and 3.2.4, or

    b.   Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate and DNBR Margin shall be determined to be within the limits of Specifications 3.2.1 and 3.2.4, respectively, by monitoring them continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEA's to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10/3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT AND REGULATING CEA INSERTION LIMITS

This special test exception permits the center CEA to be misaligned or Regulating Group 6 inserted beyond the Transient Insertion Limit during PHYSICS TESTS required to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient.

DESCRIPTION OF PROPOSED CHANGE NPF-10/15-169
AND SAFETY ANALYSIS

This is a request to revise Section 5.3.1 - Design Features - Reactor Core - Fuel Assemblies of the Technical Specifications for San Onofre Nuclear Generating Station, Units 2 and 3.

Description

The proposed change revises Technical Specification 5.3.1 - Design Features - Reactor Core - Fuel Assemblies which specifies various fuel assembly design limits including the maximum total weight of uranium in a fuel rod. The inclusion of such information in the Technical Specifications meets the requirement of 10 CFR 50.36.C.4 - "Design Features" as it related to including in the Technical Specifications features such as materials of construction and geometric arrangements. The proposed change would increase the maximum total weight of uranium in a fuel rod from 1807 gm to 1900 gm. This change is required to envelope as-built variations or possible fuel density changes which may be included in future cycles.

The proposed change does not affect the maximum fuel enrichment specified in Section 5.3.1. Further, the actual uranium weight per fuel rod is explicitly accounted for in the core performance analysis and the reactor safety analysis. The maximum weight of uranium in a fuel rod during Cycle 2 operation will be approximately 1820 gm.

Existing Technical Specifications

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

Proposed Technical Specifications

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

Safety Analysis

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

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

    Response: No

    The proposed change requests an increase in the total weight in grams of uranium per fuel rod and does not affect the specified uranium enrichment. The actual uranium weight per fuel rod is explicitely accounted for in the core performance and reactor safety analyses. Therefore, the 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 method of plant operation or operating procedures, therefore, no new path is created which may lead to a new or different kind of accident.

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 requests an increase in total weight in grams of uranium per fuel rod and does not affect the allowable enrichment. Further, the actual uranium weight per fuel rod is explicitly accounted for in the core performance analysis and the reactor safety analysis. Therefore, the 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 least likely to involve significant hazards considerations. Example (iii) from the Federal Register describes a change resulting from a nuclear reactor core reloading, where no fuel assemblies are significantly different from those found previously acceptable to the NRC for a previous core at the facility in question. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and that the NRC has previously found such methods acceptable. The proposed change is similar to example (iii) in that the change is to accommodate fuel density changes which may be included in future cycles as a result of normal reactor core reloading. The fuel assemblies are not significantly different from those found previously acceptable to the NRC, nor are there any significant changes to the acceptance criteria of the Technical Secifications or the analytical methodology used to demonstrate conformance with the technical specifications and regulations.

Safety and Significant Hazards Determination

Based on the above discussion, the proposed change does not involve a significant hazards consideration in that it does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. In addition, it is concluded that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (2) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

PWS:2407F

Attachment A

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1807 grams uranium. The initial core loading shall have a maximum enrichment of 2. 91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

b. For a pressure of 2500 psia, and

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

Attachment B

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel
assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each
fuel rod shall have a nominal active fuel length of 150 inches and contain a
maximum total weight of 1900 grams uranium. The initial core loading shall
have a maximum enrichment of 2. 91 weight percent U-235. Reload fuel shall be
similar in physical design to the initial core loading and shall have a
maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

  b. For a pressure of 2500 psia, and

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

Attachment C

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel
assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each
fuel rod shall have a nominal active fuel length of 150 inches and contain a
maximum total weight of 1807 grams uranium. The initial core loading shall
have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be
similar in physical design to the initial core loading and shall have a
maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

    b.   For a pressure of 2500 psia, and

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

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Attachment D

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1900 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.7 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

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

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

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

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

    b.   For a pressure of 2500 psia, and

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

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