REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling train shall be OPERABLE and in operation,* and either:

a. One additional shutdown cooling train shall be OPERABLE,# or

b. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5# with Reactor Coolant loops filled.

ACTION:

a. With less than the above required shutdown trains/loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators, when required, shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 The shutdown cooling train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling train may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling train is OPERABLE and in operation.

*The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

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ATTACHMENT B

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 a. At least one of the following loop(s)/trains listed below shall be OPERABLE and in operation*:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant Pump**

2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant Pump**

3. Shutdown Cooling Train A

4. Shutdown Cooling Train B

b. One additional Reactor Coolant Loop/shutdown cooling train shall be OPERABLE, or

c. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5, with Reactor Coolant loops filled.

ACTION:

a. With less than the above required shutdown trains/loops OPERABLE or with less than the required steam generator level, immediately initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.

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

---

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

** A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 285°F unless (1) the pressurizer water volume is less than 900 cubic feet, or (2) the secondary water temperature of each steam generator islless than 100°F above each of the Reactor Coolant system cold leg temperatures.

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$10% (wide range) at least once per 12 hours.

4.4.1.4.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

DESCRIPTION OF PROPOSED CHANGE NPF-15-33 AND SAFETY ANALYSIS

This is a request to revise Technical Specifications 2.2.1 (Table 2.2-1), 3.2.4, 4.2.4.4 and associated bases.

DNBR MARGIN

Existing Specifications and Bases

See Attachment A

Proposed Specifications and Bases

See Attachment B for Table 2.2-1, 4.2.4.4 and associated bases.

Reason for Proposed Change

The proposed change to Note 5 in Table 2.2-1 of Technical Specification 2.2.1 removes the specific methodology to calculate the minimum DNBR trip setpoint from the safety system settings. The methodology and references to reports containing values which may change for future cycles is more appropriately addressed in the bases section of the Technical Specifications. The proposed change to bases section, DNBR - Low, describes the specific methodology used by CPC and COLSS to determine the minimum DNBR trip setpoint. A discussion has also been added as to how the DNBR rod bow penalties of Technical Specification 4.2.4.4 are applied to the minimum DNBR trip setpoint calculation.

The proposed change to the burnup dependent penalty multiplier's in Technical Specification 4.2.4.4 is based on NRC approval of rod bow calculational methodology in Combustion Engineering topical report CENPD-225. The reduced rod bow penalties are expected to be applicable to SONGS Unit 2.

The surveillance requirements of Technical Specification 4.2.4.4 are based on burnup dependence and therefore it is appropriate to express the interval in effective full power days instead of calendar days. This change will conform with other Technical Specifications based on fuel exposure. Flexibility is added in the proposed change so that a more conservative value may be used prior to experiencing a step change in the DNBR penalty as fuel exposure increases.

Wording is added to clarify that the surveillance verification is performed indirectly on the addressable constants BERR1 for CPC and EPOL2 for COLSS. Burnup dependent penalty multipliers are used on the addressable constants to include the effects of DNBR rod bow penalties on the minimum DNBR trip setpoint calculation. The note to the DNBR penalty table in Technical Specification 4.2.4.4 has been deleted. A paragraph has been added to the bases section 3/4 2.4, DNBR MARGIN, to discuss the DNBR rod bow penalties. The description of an alternate calculational method was dropped since it is not available on CPC or COLSS.

Safety Analysis

The burnup dependent DNBR rod bow penalties proposed for Technical Specification 4.2.4.4 are based on NRC approval of Combustion Engineering topical report CENPD-225. The rod bow penalty methodology described in this report has been applied to Arkansas Nuclear One (ANO) Unit 2. Differences in the fuel spacer grids between ANO Unit 2 and San Onofre Unit 2 are accounted for in the minimum DNBR limit calculation. Therefore, the proposed DNBR rod bow penalties are expected to be applicable to San Onofre Unit 2.

The surveillance intervals of proposed Technical Specification 4.2.4.4 is changed from 31 days to 31 effective full power days. Since the surveillance is based on fuel exposure, a possible increase in the interval will not affect the safety analysis. This change in interval conforms to other Technical Specifications which are based on fuel exposure.

Allowing a more conservative value to be used prior to experiencing a step change in the DNBR rod bow penalty will increase the minimum allowed DNBR safety trip setpoint. This will add margin to the DNBR in a conservative manner.

The other changes proposed are editorial in nature and seek to clarify the Technical Specifications or add to the discussion of the bases.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-33 does not present significant hazard consideratons not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

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 is 1.20. A DNBR trip setpoint of 1.19 is allowed provided that the difference is compensated by an increase in the addressable constant BERR1. The minimum allowable value of BERR1 is 1.15 before DNBR compensation. The BERR1 adjustment shall be

$$BERR1_{NEW} = BERR1_{OLD}\left[1+\Delta DNBR(\%)\times 0.01\times \frac{d(\%POL)}{d(\%DNBR)}\right]$$

where ΔDNBR(%) is the percent increase in DNBR trip setpoint requirement and d(%POL)/d(%DNBR) is the absolute value of the most adverse derivative of percent POL with respect to percent DNBR as reported in CEN-184(S)-P.

(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.

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	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator goes below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

Seismic - High

The Seismic - High trip is provided to trip the reactor in the event of an earthquake which exceeds 60% of the Safe Shutdown Earthquake level. This trip's setpoint does not correspond to a safety limit and no credit was taken in the accident analyses for operation of this trip.

Loss of Load

The Loss of Load trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

Burnup $(\frac{GWD}{MTU})$	DNBR Penalty (%)*
0-2.4	0.0
2.4-5	3.0
5-10	7.1
10-15	10.3
15-20	12.9
20-25	15.3
25-30	17.4
30-35	19.4
35-40	21.2

*The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches. An alternate method is to determine the penalty for each individual assembly in the core based on that assembly's burnup and apply that penalty to that assembly's radial power peak.

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AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

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

3/4.2.4 DNBR MARGIN

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

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

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

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ATTACHMENT B

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 1.2.0. 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) 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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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	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
INSERT A → i.	Quality Margin-Low	< 0

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator goes below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

Seismic - High

The Seismic - High trip is provided to trip the reactor in the event of an earthquake which exceeds 60% of the Safe Shutdown Earthquake level. This trip's setpoint does not correspond to a safety limit and no credit was taken in the accident analyses for operation of this trip.

Loss of Load

The Loss of Load trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting enhances the overall reliability of the Reactor Protection System.

INSERT A

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.

This illustrates the methodology used for conversion of any DNBR penalty into a format that is useable and addressable in both CPC and COLSS. The addressable constants BERR1 and EPOL2 are also used to accommodate the DNBR rod bow penalties listed in Technical Specification 4.2.4.4.

AZIMUTHAL POWER TILT - T_q (Continued)

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

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

3/4.2.4 DNBR MARGIN

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

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

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

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INSERT A →

Insert A

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

DESCRIPTION OF PROPOSED CHANGE NPF-15-36 AND SAFETY ANALYSIS

This is a request to revise Appendix "A" Technical Specification 3.3.2 (Table 3.3-5) and 4.7.1.2.1.a.

Existing Specification

See Attachment A

Proposed Specification

A. Specification 3.3.2 (Table 3.3-5) ENGINEERED SAFETY FEATURES RESPONSE TIMES

1. Item 2(1):

(1) Safety Injection

(a)	High Pressure Safety Injection	31.2*
(b)	Low Pressure Safety Injection	41.2*
(c)	Charging Pumps	31.2*

REASON FOR PROPOSED CHANGE: Charging flow is required on pressurizer pressure-low (only) to augment HPSI flow for small break LOCA (hence, the identical response time requirement).

2. Item 3.b:

b. CIAS

(1)	Containment Isolation	10.9*	(NOTE 2)
(2)	Main Feedwater Backup Isolation (HV1105, HV1106, HV4047, HV4051)	10.9	

REASON FOR PROPOSED CHANGE: Main feedwater backup isolation valves are required to isolate main feedwater in the event of a main steam or feedline break inside containment with concurrent single failure of a MFWIV (with the identical response time requirement to MFWIV's).

3. Item 5:

MSIS

(1)	Main Steam Isolation (HV8204, HV8205)	5.9
(2)	Main Feedwater Isolation (HV4048, HV4052)	10.9
(3)	Steam, Blowdown, Sample and Drain Isolation (HV8200, HV8419, HV4054, HV4058, HV8203, HV8248) (HV8201, HV8421, HV4053, HV4057, HV8202, HV8249)	20.9

(4)	Auxiliary Feedwater Isolation (HV4705, HV4713, HV4730, HV4731) (HV4706, HV4712, HV4714, HV4715)	40.9

REASON: To clarify isolation time requirements for the various MSIS-actuated valves and correct MSIV response time.

4. Items 8 and 9

EFAS

(1)	Auxiliary Feedwater (AC trains)	52.7*/42.7**
(2)	Auxiliary Feedwater (Steam/DC train)	42.7 (NOTE 6)

REASON FOR PROPOSED CHANGE: To increase response time requirements up to the analyzed limits for AFW delivery (42.7 seconds for non-LOCA events [bounded by the loss of normal feedwater event] and 52.7 seconds for events which require AFW with SIAS present [bounded by the (coincident) loss of normal A/C event (53 seconds vs. the requested 52.7)]).

B. Specification 4.7.1.2.1.a, AUXILIARY FEEDWATER SYSTEM, add new surveillance requirement

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

REASON FOR PROPOSED CHANGE: This change is required to support changes to Table 3.3-5 Items 8 and 9. Changes to Table 3.3-5 Items 8 and 9 are based on analysis limits which are for AFW delivery versus pump start/valve stroke time. Lines are long enough that system transport time could result in unacceptable delivery time, if less than completely filled, even though pumps and valves meet the Item 8 and 9 changed requirements.

Safety Analysis

The proposed changes restrict the response time of active system components and require that system remained filled to eliminate fluid transport time in order to ensure that overall Auxiliary Feedwater System response time is within the limits of existing safety analyses. Accordingly, it is concluded that: (1) Proposed Change NPF-15-36 does not present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(c)	
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (NOTE 4a)	21.2
(c) CCW Valves (NOTE 4b)	23.2*
(d) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (NOTE 4a)	21.0
(c) CCW Valves (NOTE 4b)	23.0*
(d) Emergency Cooling Fans	21.0*
b. CIAS	
Containment Isolation	10.9* (NOTE 2)
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

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Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
MSIS	
(1) Main Steam Isolation (MSIV)	20.9
(2) Main Feedwater Isolation	10.9
6. Refueling Water Storage Tank - Low	
RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Valves Shut	50.7*
7. 4.16 kV Emergency Bus Undervoltage	
LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (steam/DC train)	30.9 (NOTE 6)
9. Steam Generator Level - Low (and ΔP - High)	
EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (Steam/DC train)	30.9 (NOTE 6)
10. Control Room Ventilation Airborne Radiation	
CRIS	
Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
TGIS	
Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

2. Verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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

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DESCRIPTION OF PROPOSED CHANGE NPF-15-43 AND SAFETY ANALYSIS

This is a request to revise Appendix "A" Technical Specification 6.8.2 and 6.8.3.

Existing Specifications

See Attachment A

Proposed Specifications

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Manager, Operations, (2) the Manager, Technical, (3) the Manager, Maintenance, (4) the Deputy Station Manager, (5) the Manager, Health Physics, (6) the Manager, Station Security, (7) the Manager, Station Emergency Preparedness, (8) the Manager, Material and Administrative Services, or (9) the Manager, Configuration Control and Compliance, as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

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

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

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

c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, (5) the Manager, Health Physics, (6) the Manager, Station Security, (7) the Manager, Station Emergency Preparedness, (8) the Manager, Material and Administrative Services, or (9) the Manager, Configuration Control and Compliance, as previously designated by the Station Manager; within 14 days of implementation.

Reason for Proposed Change

As presently written, the Station Manager may designate the appropriate functional managers to review and approve procedures in their respective areas of responsibility. In order to establish consistent requirements for review and approval in all Station Departments, the Manager of Station Security, Emergency Preparedness, Material and Administrative Services, and Configuration Control and Compliance should be included in specifications 6.8.2 and 6.8.3.

Safety Analysis

The proposed change authorizes functional managers of the Station staff to review and approve Station procedures and changes thereto that govern activities in their respective areas of responsibility. These managers are qualified, by virtue of their education, experience, and training to provide this function. The Station Manager retains overall responsibility for procedures as required by 6.5.2.1. Accordingly, it is concluded that: (1) Proposed change NPF-15-43 does not present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

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

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

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

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

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

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

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

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

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

a. Primary Coolant Sources Outside Containment

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

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

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

*See Specification 6.13.1

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g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

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

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

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

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

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

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

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

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

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

a. Primary Coolant Sources Outside Containment

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

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

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

*See Specification 6.13.1

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DESCRIPTION OF PROPOSED CHANGE NPF-15-44 AND SAFETY ANALYSIS

This is a request to revise Appendix "A" Technical Specification 6.12.

Existing Specification

See Attachment A

Proposed Specification

6.12.3 During periods when the reactor is in Modes 1, 2, 3, and 4 and when the reactor is in Mode 5 for less than 15 consecutive days, posting and locking (or otherwise securing) the access points to reactor containment may be substituted for the requirements for posting and barricading in 6.12.1 and the requirements for posting, locking, roping off, and providing flashing red lights in 6.12.2. Access to the reactor containment is permitted with the approval of the Shift Supervisor on duty and/or Health Physics Supervisor.

Reason for Proposed Change

The proposed change clarifies the requirement regarding access control to high radiation areas inside containment during operation in modes requiring containment integrity. Region V inspectors have requested such clarification.

Safety Analysis

Containment access control during Modes 1 through 4 and following initial entry into Mode 5, provides adequate control of access to high radiation areas inside containment without having to barricade, post, lock, rope off such areas or to provide flashing lights as warning against entry into such areas. Accordingly, it is concluded that: (1) Proposed Change NPF-15-44 does not present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change, and (3) this action will not result in a condition which significantly alters the impact of the Station on the environment as described in the NRC Final Environmental Statement.

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h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRC and the NSG.

l. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the REP issuance requirement during the performance of their assigned radiation protection duties, provided they are otherwise following approved plant radiation protection procedures for entry into high radiation areas.

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c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Exposure Permit.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.#

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the OSRC.

2. Shall become effective upon review and acceptance by the OSRC.

**Measurement made at 18" from source of radioactivity.
#The PCP shall be submitted and approved prior to shipment of "wet" solid radioactive waste.

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DESCRIPTION OF PROPOSED CHANGE NPF-15-60 AND SAFETY ANALYSIS
OPERATING LICENSE NPF-15
SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 3

This is a request to revise Appendix "A" Technical Specification 3.3.2, Table 3.3-5. This request deletes Items 2.a.(5)(b) and 3.a.(4)(b) and replaces them with new Item 3.b.(3), and modifies NOTE 3.

Exiting Specification

See Attachment A.

Proposed Specification

See Attachment B.

Reason for Proposed Change

This change removes the Pressurizer Pressure-Low actuation from Component Cooling Water (CCW) non-critical loop containment isolation valves HV-6211 and HV-6216 and CCW critical/non-critical loop isolation valves HV6212, HV6213, HV6218 and HV6219. These valves will continue to be actuated by Containment Pressure-High (and for the critical/non-critical loop isolation valves, low-low CCW surge tank level) in order for the CCW system to continue to respond correctly to high energy events within containment. Removal of the Pressurizer Pressure-Low signal to these valves will permit the CCW system to continue cooling non-critical loop loads (primarily the Reactor Coolant Pump (RCP) motors and seals, and the Control Element Drive Mechanism (CEDM) windings) during events which do not otherwise require isolation of the non-critical loop. For the RCP motors and seals and the CEDM windings, this change will minimize cumulative damage experienced due to loss of cooling; for the RCP seals, this change will thereby minimize the possibility of excessive RCS leakage resulting from seal failure.

Safety Analysis

The proposed change described above will remove the Pressurizer Pressure-Low actuation from the CCW non-critical loop containment isolation valves and CCW critical/non-critical loop isolation valves. These valves will continue to be actuated by Containment Pressure-High (and for the critical/non-critical loop isolation valves, low-low surge tank level). Consequently, the proposed change will only affect the response of the CCW system for those events which result in Pressurizer Pressure-Low but not Containment Pressure-High within the time assumed by existing analyses. Such events are pressurizer pressure control system failures, main steam or feedwater system control system or piping failures outside containment, and small steam, feedwater and reactor coolant system piping failures inside containment. The change in CCW system response to such events is discussed below:

Insofar as the effect on CCW system heat removal capability, it is noted that the non-critical loop loads are less than that of the critical loop-aligned Shutdown Cooling Heat Exchanger (SDCHX) and Containment Emergency Coolers. Since the SDCHX CCW valves do not open until

Containment Pressure-High-High, and the Containment Emergency Coolers do not have a significant heat load until sufficient energy has been blown into containment to exceed Containment Pressure-High, the CCW heat load for the post-SIAS/pre-CIAS configuration is much less than the design basis heat load. Although total CCW flow is higher than normal in this combined configuration, and pump head and hence per-component flow about 10% lower, the much lower than design basis heat loads result in sufficiently lower CCW temperatures to more than offset the slightly reduced flow. Hence, there is no reduction of CCW critical loop heat removal capability.

Insofar as the effect on CCW system integrity, all CCW non-critical loop piping which is impacted by reactor coolant, steam or feedwater system piping in the small break range (i.e., 2 inch through 16 inch piping ruptures which could result in Pressurizer Pressure-Low but not Containment Pressure-High within the time assumed by existing analyses) have a larger diameter than the initiating pipe; consequently, damage to CCW non-critical loop piping will be limited to moderate energy cracks rather than complete ruptures. Previous analyses have demonstrated that isolation of the non-critical loop on low-low surge tank level (a 1E signal) will protect the associated critical loop for moderate energy cracks in non-critical loop piping.

Removal of Pressurizer Pressure-Low actuation from the CCW non-critical loop containment isolation valves and CCW critical/non-critical loop isolation valves does reduce the diversity of actuation for these components; however, removal of this signal will eliminate cumulative damage to RCP seals and motors due to loss of cooling, during events which do not otherwise require isolation of the CCW non-critical loop. Minimizing such cumulative damage will increase the availability of RCPs for non-LOCA events and reduce the probability of excessive RCS leakage resulting from RCP seal failure. Further, the above discussion demonstrates acceptable consequences without Pressurizer Pressure-Low actuation of these valves.

Accordingly, it is concluded that: (1) Propose Change NPF-15-60 does not present signifcant hazard considerations not descibed or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which signifcantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

(Existing Specification)

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (NOTE 4a)	21.2
(c) CCW Valves (NOTE 4b)	23.2*
(d) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (NOTE 4a)	21.0
(c) CCW Valves (NOTE 4b)	23.0*
(d) Emergency Cooling Fans	21.0*
b. CIAS	
Containment Isolation	10.9* (NOTE 2)
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

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Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
TGIS	
Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
FHIS	
Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
CPIS	
Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.
2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to Table 3.6-1 for containment isolation valve closure times.
3. All CIAS-Actuated valves except MSIVs and MFIVs.

4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372 and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.
6. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 sec.) is included.

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ATTACHMENT B

(Proposed Specification)

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
a. (1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (Note 4b)	23.2*
(c) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (Note 4b)	23.0*
(c) Emergency Cooling Fans	21.0*
b. CIAS	
(1) Containment Isolation	10.9* (NOTE 2)
(2) [See Proposed Change No. NPF-10-36]	
(3) CCW Valves (Note 4a)	20.9
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

Table 3.3-5 (Continued)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.

2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to table 3.6-1 for containment isolation valve closure times.

3. All CIAS-Actuated valves except MSIVs, MFIVs and CCW Valves 3HV-6211 and 3HV-6216.

4a. CCW non-critical loop isolation valves 3HV-6212, 3HV-6213, 3HV-6218 and 3HV-6219 close.

4b. Containment emergency cooler CCW isolation valves 3HV-6366, 3HV-6367, 3HV-6368, 3HV-6369, 3HV-6370, 3HV-6371, 3HV-6372, and 3HV-6373 open.

5. Response time includes instrumentation, logic, and isolation damper closure times only.

6. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3.

* Emergency diesel generator starting delay (10 seconds) and sequence loading delays for SIAS are included.

** Emergency diesel generator starting delay (10 seconds) is included.

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