ENCLOSURE 1

Consumers Power Company
Palisades Plant
Docket 50-255

TECHNICAL SPECIFICATIONS CHANGE REQUEST

Revised Technical Specification Pages

February 3, 1992

6 Pages

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Reactor Core

The Minimum DNBR of the reactor core shall be maintained greater than or equal to the DNB correlation safety limit.

Correlation	Safety Limit
XNB	1.17
ANFP	1.154

Applicability

Safety Limit 2.1 is applicable in HOT STANDBY and POWER OPERATION.

Action

2.1.1 If a Safety Limit is exceeded, comply with the requirements of Specification 6.7

2.2 Safety Limit - Primary Coolant System Pressure (PCS)

The PCS Pressure shall not exceed 2750 psia.

Applicability

Safety Limit 2.2 is applicable when there is fuel in the reactor.

Action

2.2.1 If a Safety Limit is exceeded, comply with the requirements of Specification 6.7

2.3 Limiting Safety System Settings - Reactor Protective System (RPS)

The RPS trip setting limits shall be as stated in Table 2.3.1.

Applicability

Limiting Safety System Settings of Table 2.3.1 are applicable when the associated RPS channels are required to be OPERABLE by Specification 3.17.1.

Action

2.3.1 If an RPS instrument setting is not within the allowable settings of Table 2.3.1, immediately declare the instrument inoperable and complete corrective action as directed by Specification 3.17.1.

TABLE 2.3.1

REACTOR PROTECTIVE SYSTEM TRIP SETTING LIMITS

	RPS Trip Unit	Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating
1.	Variable High Power	$\leq$15% above core power, with a minimum of $\leq$30% RATED POWER and a maximum of $\leq$106.5% RATED POWER.	$\leq$15% above core power with a minimum of $\leq$15% RATED POWER and a maximum of $\leq$49% RATED POWER.
2.	PCS Flow	$\geq$95% Full PCS Flow.	$\geq$60% Full PCS Flow.
3.	High Pressure Pressurizer	$\leq$2255 psia.	$\leq$2255 psia.
4.	Thermal Margin/ Low Pressure	(a)	(a)
5.	Steam Generator Low Water Level	Above the feedwater ring center line.	Above the feedwater ring center line.
6.	Steam Generator Low Pressure	$\geq$500 psia.	$\geq$500 psia.
7.	Containment High Pressure	$\leq$3.70 psig.	$\leq$3.70 psig.

(a) The pressure setpoint for the Thermal Margin/Low Pressure Trip, P_{trip}, is the higher of two values, P_{min} and P_{var}, both in psia:

$$P_{min} = 1750$$
$$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9493$$

where:

$QA = -0.720(ASI) + 1.028;$	when $-0.628 \leq ASI < -0.100$
$QA = -0.333(ASI) + 1.067;$	when $-0.100 \leq ASI < +0.200$
$QA = +0.375(ASI) + 0.925;$	when $+0.200 \leq ASI \leq +0.565$
ASI = Measured ASI	when $Q \geq 0.0625$
$ASI = 0.0$	when $Q < 0.0625$
$QR_1 = 0.412(Q) + 0.588;$	when $Q \leq 1.0$
$QR_1 = Q;$	when $Q > 1.0$

Q = Core Power/Rated Power

T_{in} = Maximum primary coolant inlet temperature, in °F.

ASI, T_{in}, and Q are the existing values as measured by the associated instrument channel.

Amendment No. 31, 80, 118, 138,

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System

	Channel Description	Surveillance Function	Frequency	Surveillance Method
1.	Power Range Safety Channels	a. Check[7]	S	a. Comparison of four-power channel readings.
		b. Check[3]	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparators alarms.
		c. Test	M[2]	c. Internal test signal.
		d. Calibrate[6]	R	d. Channel alignment through measurement/adjustment of internal test points.
2.	Wide-Range Neutron Monitors	a. Check	S	a. Comparison of channel indications.
		b. Test	P	b. Internal test signal.
		c. Calibrate	R	c. Channel alignment through measurement/adjustment of internal test points.
3.	Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.
		b. Calibrate	R	b. Known differential pressure applied to sensors.
		c. Test	M[2]	c. Bistable trip tester.[1]
4.	Thermal Margin/Low Pressurizer Pressure	a. Check:[8]	S	a. Check:
		(1) Temperature input		(1) Comparison of four separate calculated trip pressure set point indications.
		(2) Pressure input		(2) Comparison of four pressurizer pressure indications. Same as 5(a) below.)
		b. Calibrate	R	b. Calibrate:
		(1) Temperature input		(1) Known resistance substituted for RTD coincident with known pressure and power input.
		(2) Pressure input		(2) Part of 5(b) below.
		c. Test	M[2]	c. Bistable trip tester.[1]
5.	High-Pressurizer Pressure	a. Check [8]	S	a. Comparison of four separate pressure indications.
		b. Calibrate	R	b. Known pressure applied to sensors.
		c. Test	M[2]	c. Bistable trip tester.[1]

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System (continued)

	Channel Description	Surveillance Function	Frequency	Surveillance Method
6.	Steam Generator Level	a. Check	S	a. Comparison of four level indications per generator.
		b. Calibrate	R	b. Known differential pressure applied to sensors.
		c. Test	M(2)	c. Bistable trip tester.(1)
7.	Steam Generator Pressure	a. Check	S	a. Comparisons of four pressure indications per generator
		b. Calibrate	R	b. Known pressure applied to sensors.
		c. Test	M(2)	c. Bistable trip tester.(1)
8.	Containment Pressure	a. Calibrate	R	a. Known pressure applied to sensors.
		b. Test	M(2)	b. Simulate pressure switch action.
9.	Loss of Load	a. Test	P	a. Manually trip turbine auto stop oil relays.
10.	Manual Trips	a. Test	P	a. Manually test both circuits.
11.	Reactor Protection System Logic Units	a. Test	M(2)	a. Internal test circuits.
12.	Axial Shape Index (ASI)	a. Test	R	a. Known power inputs applied to Thermal Margin Calculator.
13.	ΔT Power	a. Check(7)	S	a. Same as 1(a).
		b. Check(3)	D	b. Same as 1(b).
		c. Test	R	c. Known temperature inputs applied to Thermal Margin Calculator.

Amendment No ~~30~~, ~~66~~, ~~118~~, 130
March 23, 1990

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System (continued)

	Channel Description	Surveillance Function	Frequency	Surveillance Method
14.	Thermal Margin Calculator	a. Check	Q	a. Verify constants.

NOTES:

(1) The bistable trip tester injects a signal into the bistable and provides a precision readout of the trip set point.

(2) All monthly tests will be done on only one of four channels at a time to prevent reactor trip.

(3) Adjust the nuclear power or ΔT power until readout agrees with heat balance calculations when above 15% of rated power.

(4) Deleted

(5) It is not necessary to perform the specified testing during prolonged periods in the refueling shutdown condition If this occurs, omitted testing will be performed prior to returning the plant to service.

(6) Also includes testing variable high power function in the Thermal Margin Calculator.

(7) Required if the reactor is critical.

(8) Required when PCS is >1500 psia.

FREQUENCY Notation

Notation	Frequency
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
P	Prior to each start-up if not done previous week.
NA	Not applicable.

ENCLOSURE 2

Consumers Power Company
Palisades Plant
Docket 50-255

TECHNICAL SPECIFICATIONS BASIS REVISION

Revised Limiting Safety System Settings Basis pages

February 3, 1992

5 Pages

2.0 BASIS - Safety Limits and Limiting Safety System Settings

2.1 Basis - Reactor Core Safety limit

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high-cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of thermal power, primary coolant flow, temperature and pressure, can be related to DNB through the use of a DNB Correlation. DNB Correlations have been developed to predict DNB and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to DNB correlation safety limit. A DNBR equal to the DNB correlation safety limit corresponds to a 95% probability at a 95% confidence level that DNB will not occur which is considered an appropriate margin to DNB for all operating conditions.

The reactor protective system is designed to prevent any anticipated combination of transient conditions for primary coolant system temperature, pressure and thermal power level that would result in a DNBR of less than the DNB correlation safety limit. The Palisades safety analyses uses two DNB correlations. The XNB correlation discussed in References 1 and 2 determines the safety limit for those fuel assemblies initially loaded in Cycle 8. The ANFP correlation discussed in References 4 and 5 determines the safety limit for those fuel assemblies initially loaded in Cycle 9 and later. Fuel assemblies initially loaded in Cycle 8 are of a different construction than later assemblies which utilize a High Thermal Performance design.

The minimum DNBR analyses are in accordance with Reference 6.

References

(1) XN-NF-621(P)(A), Rev 1
(2) XN-NF-709
(3) Updated FSAR, Section 14.1.
(4) ANF-1224 (P)(A), May 1989
(5) ANF-89-192(P), January 1990
(6) XN-NF-82-21(A), Revision 1

Amendment No. ~~31~~, ~~43~~, ~~118~~, ~~137~~,

2.2 Basis - Primary Coolant System Safety Limit

The primary coolant system[1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The Primary Coolant System design pressure is 2500 psia. The maximum allowable Primary Coolant System transient pressure is limited by the pressure vessel limit (ASME Code, Section III) of 110% of design pressure and by the piping, valve, and fitting limit (ASA Section B31.1) of 120% of design pressure. The initial hydrostatic test was conducted at 125% of design pressure (3125 psia) to verify the integrity of the primary coolant system. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings of the reactor High Pressure Trip, primary safety valves, and secondary safety valves have been established to assure never reaching the primary coolant system safety limit. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by the normal setting of the atmospheric steam dump and turbine bypass valves of 900 psia.

References

(1) Updated FSAR, Section 4.
(2) Updated FSAR, Section 4.3.

2.3 Basis - Limiting Safety System Settings

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

1. Variable High Power - The Variable High Power Trip (VHPT) is incorporated in the reactor protection system to provide a reactor trip for transients exhibiting a core power increase starting from any initial power level (such as the boron dilution transient). The VHPT system provides a trip setpoint no more than a predetermined amount above the indicated core power with a specified upper limit. Operator action is required to increase the setpoint as core power is increased; the setpoint is automatically decreased as core power decreases. Provisions have been made to select different set points for three pump and four pump operations.

   During normal plant operation with all primary coolant pumps operating, reactor trip is initiated when the reactor power level reaches 106.5% of indicated rated power. Adding to this the possible variation in trip point due to calibration and instrument errors, the maximum actual steady state power at which a trip would be actuated is 115%, which was used for the purpose of safety analysis.[5]

2. Primary Coolant System (PCS) Low Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.[2] Flow in each of the four coolant loops is determined from pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is determined, for the RPS flow channels, by summing the loop pressure drops across the steam generators and correlating this pressure sum with the sum of steam generator differential pressures which exists at 100% flow (four pump operation at full power t_{ave}). The normal flow with three pumps operating is 74.7% of Full PCS Flow. Full PCS flow is that flow which exists at RATED POWER, at full power T_{ave}, with four pumps operating.

   During four pump operation, the Low Flow Trip setting of 95% insures that the reactor cannot operate when the flow rate is less than 93% of the nominal value considering instrument errors.[5]

   Provisions are made in the reactor protective system to permit operation of the reactor at reduced power if one coolant pump is taken out of service. These low-flow and high-flux settings have been derived in consideration of instrument errors and response times of equipment involved to assure that thermal margin and flow stability will be maintained during normal operation and anticipated transients.[4] For reactor operation with one coolant pump inoperative, core power must be reduced and then the Variable High Power and Low Flow setpoints must be adjusted to the three pump values before the pump may be stopped.

Amendment No. ~~31~~, ~~118~~, ~~137~~,

2.3 Basis - Limiting Safety System Settings (continued)

3. High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the primary and secondary safety valves to prevent primary system overpressure (Specification 3.1.7). In the event of loss of load without reactor trip, the temperature and pressure of the primary coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. This setting is consistent with the trip point assumed in the accident analysis.[8]

4. Thermal Margin/Low Pressure (TM/LP) Trip

The TM/LP trip system monitors core power, reactor coolant maximum inlet temperature, (T_{in}), core coolant system pressure and axial shape index. The Low Pressure Trip limit (P_{var}) is calculated using the equations given in Table 2.3.1.

The calculated limit (P_{var}) is then compared to a fixed Low Pressure Trip limit (p_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured PCS pressure and a trip signal is generated when the measured pressure for that channel is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting P_{trip} + ΔP.

The TM/LP trip set points are derived from the 4-pump operation core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. A pressure allowance of 165 psi is assumed to account for instrument drift in both power and inlet temperatures, calorimetric power measurement, inlet temperature measurement, and primary system pressure measurement. Uncertainties accounted for that are not a part of the 165 psi term include allowances for assembly power tilt, fuel pellet manufacturing tolerances, core flow measurement uncertainty and core bypass flow, inlet temperature measurement time delays, and ASI measurement. Each of these allowances and uncertainties are included in the development of the TM/LP trip set point used in the accident analysis.

Amendment No ~~31~~, ~~82~~, ~~118~~,

2.3 Basis - Limiting Safety System Settings (continued)

5. Low Steam Generator Water Level - The low steam generator water level reactor trip protects against the loss of feed-water flow accidents and assures that the design pressure of the primary coolant system will not be exceeded. The specified set point assures that there will be sufficient water inventory in the steam generator at the time of trip to allow a safe and orderly plant shutdown and to prevent steam generator dryout assuming minimum auxiliary feedwater capacity.[6]

   The setting listed in Table 2.3.1 assures that the heat transfer surface (tubes) is covered with water when the reactor is critical.

6. Low Steam Generator Pressure - A reactor trip on low steam generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the primary coolant. The setting of 500 psia is sufficiently below the rated load operating point of 739 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used in the accident analysis.[5]

7. Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shutdown before the initiation of the safety injection system and containment spray.[7]

References

(1) EMF-91-176, Table 15.0.7-1
(2) Updated FSAR, Section 7.2.3.3.
(3) EMF-91-176, Section 15.0.7-1
(4) XN-NF-86-91(P)
(5) ANF-90-078, Section 15.1.5
(6) ANF-87-150(NP), Volume 2, Section 15.2.7
(7) Updated FSAR, Section 7.2.3.9.
(8) ANF-90-078, Section 15.2.1

ENCLOSURE 3

Consumers Power Company
Palisades Plant
Docket 50-255

TECHNICAL SPECIFICATIONS CHANGE and BASIS REVISION

Existing Pages Marked to Show Changes

February 3, 1992

12 Pages

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS - REACTOR CORE

Applicability

SAFETY LIMIT 2.1 IS APPLICABLE ~~This specification applies when the reactor is~~ in hot standby CAPS ~~condition~~ and power operation CAPS ~~condition.~~

~~Objective~~

~~To maintain the integrity of the fuel cladding and prevent the release of significant amounts of fission products to the primary coolant.~~

Specifications

The MINIMUM DNBR of the reactor core shall be maintained greater than or equal to the DNB correlation safety limit.

Basis

BASIS MOVED TO END OF SECTION 2.

To maintain the integrity of the fuel cladding and prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high-cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of thermal power, primary coolant flow, temperature and pressure, can be related to DNB through the use of a DNB Correlation. DNB Correlations have been developed to predict DNB and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to DNB correlation safety limit. A DNBR equal to the DNB correlation safety limit corresponds to a 95% probability at a 95% confidence level that

2.1 SAFETY LIMITS - REACTOR CORE (Contd)

DNB will not occur which is considered an appropriate margin to DNB for all operating conditions.

The reactor protective system is designed to prevent any anticipated combination of transient conditions for primary coolant system temperature, pressure and thermal power level that would result in a DNBR of less than the DNB correlation safety limit. The DNB correlations used in the Palisades safety analysis are listed in the following table.

INCLUDED IN SAFETY LIMIT

Correlation ~~Name~~	Safety Limit	~~References Correlation~~	~~References Applicability~~
XNB	1.17	~~1~~	~~2~~
ANFP	1.154	~~4~~	~~5~~

The MDNBR analyses are performed in accordance with Reference 6.

References

(1) XN-NF-621(P)(A), Rev 1
(2) XN-NF-709
(3) Updated FSAR, Section 14.1.
(4) ANF-1224 (P)(A), May 1989
(5) ANF-89-192(P), January 1990
(6) XN-NF-82-21(A), Revision 1

2.2 SAFETY LIMITS - PRIMARY COOLANT SYSTEM PRESSURE

Applicability

~~Applies to the limit on primary coolant system pressure.~~

~~Objective~~

~~To maintain the integrity of the primary coolant system and to prevent the release of significant amounts of fission product activity to the primary coolant.~~

Specification

PCS

The ~~primary coolant system~~ pressure shall not exceed 2750 psia when there are fuel assemblies in the reactor vessel.

MOVED TO APPLICABILITY

Basis

BASIS MOVED TO END OF SECTION 2 AND REVISED TO REMOVE SETPOINTS FROM OTHER SPECIFICATIONS

The primary coolant system [1] serves as a barrier to prevent radionuclides in the primary coolant from reaching the atmosphere. In the event of a fuel cladding failure, the primary coolant system is the foremost barrier against the release of fission products. Establishing a system pressure limit helps to assure the continued integrity of both the primary coolant system and the fuel cladding. The maximum transient pressure allowable in the primary coolant system pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the primary coolant system piping, valves and fittings under ASA Section B31.1 is 120% of design pressure. Thus, the safety limit of 2750 psia (110% of the 2500 psia design pressure) has been established.[2] The settings and capacity of the secondary coolant system safety valves (985-1025 psig)[3], the reactor high-pressure trip (≤2400 psia) and the primary safety valves (2500-2580 psia)[4] have been established to assure never reaching the primary coolant system pressure safety limit. The initial hydrostatic test was conducted at 3125 psia (125% of design pressure) to verify the integrity of the primary coolant system. Additional assurance that the nuclear steam supply system (NSSS) pressure does not exceed the safety limit is provided by setting the secondary coolant system steam dump and bypass valves at 900 psia.

References

(1) Updated FSAR, Section 4.
(2) Updated FSAR, Section 4.3.
(3) Updated FSAR, Table 4-5
(4) Updated FSAR, Table 4-10

Applicability (NEW APPLICABILITY IS BELOW)

~~This specification applies to reactor trip settings and bypasses for instrument channels.~~ e

~~Objective~~

~~To provide for automatic protective action in the event that the principal process variables approach a safety limit.~~ e

Specification

The ~~reactor protective system~~ RPS trip setting limits ~~and the permissible bypasses for the instrument channels~~ shall be as stated in Table 2.3.1. e

MOVED TO BASIS

The TM/LP trip system monitors core power, reactor coolant maximum inlet temperature, (T_{in}), core coolant system pressure and axial shape index. The low pressure trip limit (P_{var}) is calculated using the following equation.

MOVED TO FOLLOW TABLE 2.3.1. CYCLE 10 NUMBERS USED

$$P_{var} = 2012(QA)(QR_1) + 17.0(T_{in}) - 9493$$

where:

$$QR_1 = 0.412(Q) + 0.588 \quad Q \le 1.0 \qquad Q = \frac{\text{core power}}{\text{rated power}}$$
$$= Q \quad Q > 1.0$$

$$QA = -0.720(ASI) + 1.028 \quad -0.628 \le ASI < -0.100$$
$$= -0.333(ASI) + 1.067 \quad -0.100 \le ASI < +0.200$$
$$= +0.375(ASI) + 0.925 \quad +0.200 \le ASI \le +0.565$$
$$= 1.085 \text{ when } Q < 0.0625$$

MOVED TO BASIS

The calculated limit (P_{var}) is then compared to a fixed low pressure trip limit (p_{min}). The auctioneered highest of these signals becomes the trip limit (P_{trip}). P_{trip} is compared to the measured reactor coolant pressure (P) and a trip signal is generated when P is less than or equal to P_{trip}. A pre-trip alarm is also generated when P is less than or equal to the pre-trip setting $P_{trip} + \Delta P$.

APPLICABILITY

LIMITING SAFETY SETTINGS OF TABLE 2.3.1 ARE APPLICABLE WHEN THE ASSOCIATED RPS CHANNELS ARE REQUIRED TO BE OPERABLE BY SPECIFICATION 3.17.1.

ACTION (ADDED)

SEE PROPOSED PAGE

TABLE 2.3.1

Reactor Protective System Trip Setting Limits

RPS Trip Unit	Four Primary Coolant Pumps Operating	Three Primary Coolant Pumps Operating
1. Variable High Power~~(1)~~	≤~~10%~~ 15% above core power, with a minimum ~~setpoint~~ of ≤30% ~~of~~ rated power and a maximum of ≤106.5% ~~of~~ rated power	≤~~10%~~ 15% above core power with a minimum ~~setpoint~~ ~~of~~ of ≤15% rated power and a maximum of ≤49% ~~of~~ rated power
2. PCS Flow ~~Primary Coolant Flow(2)~~	≥95% FULL PCS FLOW ~~of Primary Coolant Flow With Four Pumps Operating~~	≥60% FULL PCS FLOW ~~of Primary Coolant Flow With Four Pumps Operating~~
3. High Pressure Pressurizer	≤2255 Psia	≤2255 Psia
4. Thermal Margin/ Low Pressure~~(2,3)~~	(a) ~~P_{trip} ≥ Applicable Limits~~	(a) ~~Replaced by Variable High Power Trip and 1750 Psia Minimum Low-Pressure Setting~~
5. Steam Generator Low Water Level	~~Not Lower Than the Center Line of Feed Water Ring~~ ABOVE THE FEEDWATER RING CENTER LINE	~~Not Lower Than the Center Line of Feed Water Ring~~ ABOVE THE FEEDWATER RING CENTER LINE
6. Steam Generator Low Pressure~~(2)~~	≥500 Psia	≥500 Psia
7. Containment High Pressure	≤3.70 Psig	≤3.70 Psig

~~(1) The VHPT can be 30% of rated power for power levels ≤ 20% of rated power.~~

~~(2) May be bypassed below 10^{-4}% of rated power provided auto bypass removal circuitry is operable. For low power physics tests, thermal margin/low pressure, primary coolant flow and low steam generator pressure trips may be bypassed until their react points are reached (approximately 1750 psia and 500 psia, respectively), provided automatic bypass removal circuitry at 10^{-1}% rated power is operable.~~

~~(3) Minimum trip setting shall be 1750 psia.~~

~~(4) Operation with three pumps for a maximum of 12 hours is permitted to provide a limited time for repair/pump restart, to provide for an orderly shutdown or to provide for the conduct of reactor internals noise monitoring test measurements.~~

(a) THE PRESSURE SETPOINT FOR THE THERMAL MARGIN/LOW PRESSURE TRIP, P_{TRIP}, IS THE ~~[illegible]~~ HIGHER OF TWO VALUES, P_{MIN} AND P_{VAR}, BOTH IN PSIA:

Amendment No. 31, 80, 118, ~~138~~
~~February 22, 1991~~

EQUATIONS HERE (FORMERLY ON PAGE 2-4)

2.3 LIMITING SAFETY SYSTEM SETTINGS - REACTOR PROTECTIVE SYSTEM (Contd)

Basis

The reactor protective system consists of four instrument channels to monitor selected plant conditions which will cause a reactor trip if any of these conditions deviate from a preselected operating range to the degree that a safety limit may be reached.

Basis Editorially Revised. See Proposed Pages

1. Variable High Power - The variable high power trip (VHPT) is incorporated in the reactor protection system to provide a reactor trip for transients exhibiting a core power increase starting from any initial power level (such as the boron dilution transient). The VHPT system provides a trip setpoint no more than a predetermined amount above the indicated core power. Operator action is required to increase the setpoint as core power is increased; the setpoint is automatically decreased as core power decreases. Provisions have been made to select different set points for three pump and four pump operations.

   During normal plant operation with all primary coolant pumps operating, reactor trip is initiated when the reactor power level reaches 106.5% of indicated rated power. Adding to this the possible variation in trip point due to calibration and instrument errors, the maximum actual steady state power at which a trip would be actuated is 115%, which was used for the purpose of safety analysis.[1]

2. Primary Coolant System Low Flow - A reactor trip is provided to protect the core against DNB should the coolant flow suddenly decrease significantly.[3] Flow in each of the four coolant loops is determined from a measurement of pressure drop from inlet to outlet of the steam generators. The total flow through the reactor core is measured by summing the loop pressure drops across the steam generators and correlating this pressure sum with the pump calibration flow curves. The percent of normal core flow is shown in the following table:

4 Pumps	100.0%
3 Pumps	74.7%

   During four-pump operation, the low-flow trip setting of 95% insures that the reactor cannot operate when the flow rate is Less than 93% of the nominal value considering instrument errors.[4]

Basis (Contd)

Provisions are made in the reactor protective system to permit operation of the reactor at reduced power if one coolant pump is taken out of service. These low-flow and high-flux settings have been derived in consideration of instrument errors and response times of equipment involved to assure that thermal margin and flow stability will be maintained during normal operation and anticipated transients.[5] For reactor operation with one coolant pump inoperative, the low-flow trip points and the overpower trip points must be manually changed to the specified values for the selected pump condition by means of set point selector switches. The trip points are shown in Table 2.3.1.

3. High Pressurizer Pressure - A reactor trip for high pressurizer pressure is provided in conjunction with the primary and secondary safety valves to prevent primary system overpressure (Specification 3.1.7). In the event of loss of load without reactor trip, the temperature and pressure of the primary coolant system would increase due to the reduction in the heat removed from the coolant via the steam generators. This setting is consistent with the trip point assumed in the accident analysis.[11]

Basis (Continued)

4. Thermal Margin/Low-Pressure Trip

The TM/LP trip set points are derived from the 4-pump operation core thermal limits through application of appropriate allowances for measurement uncertainties and processing errors. A pressure allowance of 165 psi is assumed to account for: instrument drift in both power and inlet temperatures; calorimetric power measurement; inlet temperature measurement; and primary system pressure measurement. Uncertainties accounted for that are not a part of the 165 psi term include allowances for: assembly power tilt; fuel pellet manufacturing tolerances; core flow measurement uncertainty and core bypass flow; inlet temperature measurement time delays; and ASI measurement. Each of these allowances and uncertainties are included in the development of the TM/LP trip set point used in the accident analysis.

For three-pump operation, continued power operation is restricted. During this mode of operation, the high power level trip in conjunction with the TM/LP trip (minimum set point = 1750 psia) and the secondary system safety valves (set at approximately 1000 psia) assure that adequate DNB margin is maintained.[5]

5. Low Steam Generator Water Level - The low steam generator water level reactor trip protects against the loss of feed-water flow accidents and assures that the design pressure of the primary coolant system will not be exceeded. The specified set point assures that there will be sufficient water inventory in the steam generator at the time of trip to allow a safe and orderly plant shutdown and to prevent steam generator dryout assuming minimum auxiliary feedwater capacity.[9]

The setting listed in Table 2.3.1 assures that the heat transfer surface (tubes) is covered with water when the reactor is critical.

Basis (Contd)

6. Low Steam Generator Pressure - A reactor trip on low steam generator secondary pressure is provided to protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the primary coolant. The setting of 500 psia is sufficiently below the rated load operating point of 739 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used in the accident analysis.[8]

7. Containment High Pressure - A reactor trip on containment high pressure is provided to assure that the reactor is shutdown before the initiation of the safety injection system and containment spray.[10]

8. Low Power Physics Testing - For low power physics tests, certain tests will require the reactor to be critical at low temperature (≥260°F) and low pressure (≥415 psia). For these certain tests only, the thermal margin/low pressure, primary coolant flow and low steam generator pressure trips may be bypassed in order that reactor power can be increased for improved data acquisition. Special operating precautions will be in effect during these tests in accordance with approved written testing procedures. At reactor power levels below 10^{-1}% of rated power, the thermal margin/low-pressure trip and low flow trip are not required to prevent fuel rod thermal limits from being exceeded. The low steam generator pressure trip is not required because the low steam generator pressure will not allow a severe reactor cooldown, should a steam line break occur during these tests.

References

(1) ANF-90-078, Table 15.0.7-1
(2) deleted
(3) Updated FSAR, Section 7.2.3.3.
(4) ANF-90-078, Section 15.0.7-1
(5) XN-NF-86-91(P)
(6) deleted
(7) deleted
(8) ANF-90-078, Section 15.1.5
(9) ANF-87-150(NP), Volume 2, Section 15.2.7
(10) Updated FSAR, Section 7.2.3.9.
(11) ANF-90-078, Section 15.2.1

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System(S)

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Power Range Safety Channels	a. Check(7)	S	a. Comparison of four-power channel readings.
	b. Check(3)	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparators alarms.
	c. Test	M(2)	c. Internal test signal.
	d. Calibrate(6)	R	d. Channel alignment through measurement/adjustment of internal test points.
2. Wide-Range Neutron Monitors	a. Check	S	a. Comparison of channel indications.
	b. Test	P	b. Internal test signal.
	c. Calibrate	R	c. Channel alignment through measurement/adjustment of internal test points.
3. Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)~~(4)~~
4. Thermal Margin/Low Pressurizer Pressure	a. Check:(8)	S	a. Check:
	(1) Temperature Input		(1) Comparison of four separate calculated trip pressure set point indications.
	(2) Pressure Input		(2) Comparison of four pressurizer pressure indications. Same as 5(a) below.)
	b. Calibrate	R	b. Calibrate:
	(1) Temperature Input		(1) Known resistance substituted for RTD coincident with known pressure and power input.
	(2) Pressure Input		(2) Part of 5(b) below.
	c. Test	M(2)	c. Bistable trip tester.(1)
5. High-Pressurizer Pressure	a. Check (8)	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)

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No Changes to this page

TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System(5) (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
6. Steam Generator Level	a. Check	S	a. Comparison of four level indications per generator.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)
7. Steam Generator Pressure	a. Check	S	a. Comparisons of four pressure indications per generator.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M(2)	c. Bistable trip tester.(1)
8. Containment Pressure	a. Calibrate	R	a. Known pressure applied to sensors.
	b. Test	M(2)	b. Simulate pressure switch action.
9. Loss of Load	a. Test	P	a. Manually trip turbine auto stop oil relays.
10. Manual Trips	a. Test	P	a. Manually test both circuits.
11. Reactor Protection System Logic Units	a. Test	M(2)	a. Internal test circuits.
12. Axial Shape Index (ASI)	a. Test	R	a. Known power inputs applied to Thermal Margin Calculator.
13. ΔT Power	a. Check (7)	S	a. Same as 1(a).
	b. Check (3)	D	b. Same as 1(b).
	c. Test	R	c. Known temperature inputs applied to Thermal Margin Calculator.

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TABLE 4.1.1

Minimum Frequencies for Checks, Calibrations and Testing of Reactor Protective System(5) (Contd)

Channel Description	Surveillance Function	Frequency	Surveillance Method
14. Thermal Margin Calculator	a. Check	Q	a. Verify constants.

NOTES: (1)The bistable trip tester injects a signal into the bistable and provides a precision readout of the trip set point.

(2)All monthly tests will be done on only one of four channels at a time to prevent reactor trip.

(3)Adjust the nuclear power or ΔT power until readout agrees with heat balance calculations when above 15% of rated power.

(4) DELETED

~~(4)Trip setting for operating pump combination only. Settings for other than operating pump combinations must be tested during routine monthly testing performed when shut down and within four hours after resuming operation with a different pump combination if the setting for that combination has not been tested within the previous month.~~

(5)It is not necessary to perform the specified testing during prolonged periods in the refueling shutdown condition If this occurs, omitted testing will be performed prior to returning the plant to service.

(6)Also includes testing variable high power function in the Thermal Margin Calculator.

(7)Required if the reactor is critical.

(8)Required when PCS is > 1500 psia.

FREQUENCY NOTATION

Notation	Frequency
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
P	Prior to each start-up if not done previous week.
NA	Not applicable.

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