Attachment 2

PROPOSED TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

CHANNEL CALIBRATION

1.3 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds with the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.4 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL TEST

1.5 A CHANNEL TEST shall be the injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action. The CHANNEL TEST shall include adjustments, as necessary, of the alarm, interlock and/or trip setpoints, such that the setpoints are within the required range and accuracy.

CONTAINMENT INTEGRITY

1.6 CONTAINMENT INTEGRITY shall exist when:

(1) All non-automatic containment isolation valves (or blind flanges) are closed.

(2) The equipment door is properly closed.

(3) At least one door in each personnel air lock is properly closed.

(4) All automatic containment isolation valves are operable.

CORE ALTERATION

1.7 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

CORRELATION CHECK

1.8 A CORRELATION CHECK shall be an engineering analysis of an incore flux map wherein at least one point along the incore versus excore correlation data plot is obtained.

CORRELATION VERIFICATION

1.9 A CORRELATION VERIFICATION shall be the engineering analysis of incore flux maps wherein multiple points along the incore versus excore correlation data plot are obtained.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

E - AVERAGE DISINTEGRATION ENERGY

1.11 E is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant.

FIRE SUPPRESSION WATER SYSTEM

1.12 A FIRE SUPPRESSION WATER SYSTEM shall consist of a water source(s), pump(s), and distribution piping with associated isolation valves (i.e., system header, hose standpipe and spray header isolation valves).

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include nonemployees of the licensee who are permitted to use portions of the site for recreational, occupational, or purposes not associated with plant functions. This category shall not include nonemployees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.16 An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PROCESS CONTROL PROGRAM

1.19 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE-PURGING

1.20 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.21 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 1347 Mwt.

REPORTABLE EVENT

1.22 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESIDUAL HEAT REMOVAL (RHR) TRAIN

1.23 An RHR TRAIN shall be a train of components that includes: one RHR pump aligned with one RHR heat exchanger; one component cooling water pump aligned with the same RHR heat exchanger and with one component cooling water heat exchanger; and one salt water pump aligned with the same component cooling water heat exchanger.

SHUTDOWN MARGIN

1.24 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.25 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

1.26 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.27 A SOURCE CHECK is the qualitative assessment of a channel response when the channel sensor is exposed to a radioactive source.

STAGGERED TEST BASIS

1.28 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.29 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.30 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy.

UNRESTRICTED AREA

1.31 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.32 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.33 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

APPLICABILITY: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

OBJECTIVE: To maintain the integrity of the reactor coolant system and to prevent the release of excessive amounts of fission product activity to the coolant.

SPECIFICATION: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded.

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1

BASIS: Safety Limits

1. Reactor Coolant System Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the containment atmosphere. In addition, the failure of components of the Reactor Coolant System could result in damage to the fuel and pressurization of the containment. A safety limit of 2735 psig (110% of design pressure) has been established which represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section VIII.

2. Plant Operating Transients

In order to prevent any significant amount of fission products from being released from the fuel to the reactor coolant, it is necessary to prevent clad overheating both during normal operation and while undergoing system transients. Clad overheating and potential failure could occur if the heat transfer mechanism at the clad surface departs from nucleate boiling. System parameters which affect this departure from nucleate boiling (DNB) have been correlated with experimental data to provide a means of determining the probability of DNB occurrence. The ratio of the heat flux at which DNB is expected to occur for a given set of conditions to the actual heat flux experienced at a point is the DNB ratio and reflects the probability that DNB will actually occur.

It has been determined that under the most unfavorable conditions of power distribution expected during core lifetime and if a DNB ratio of 1.44 should exist, not more than 7 out of the total of 28,260 fuel rods would be expected to experience DNB. These conditions correspond to a reactor power of 125% of rated power. Thus, with the expected power distribution and peaking factors, no significant release of fission products to the reactor coolant system should occur at DNB ratios greater than 1.30.(1) The DNB ratio, although fundamental, is not an observable variable. For this reason, limits have been placed on reactor coolant temperature, flow, pressure, and power level, these being the observable process variables related to determination of the DNB ratio. The curves presented in Figure 2.1.1 represent loci of conditions at which a minimum DNB ratio of 1.30 or greater would occur. (1)(2)(3)

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less.

2. Variable Low Pressure, Loss of Flow, and Nuclear Overpower Trips

These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, control rod ejection, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)(7)

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 5%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above.

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.[3][6]

References:

(1) Amendment No. 10 to the Final Engineering Report and Safety Analysis, Section 4, Question 3

(2) Final Engineering Report and Safety Analysis, Paragraph 3.3

(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(4) Final Engineering Report and Safety Analysis, Paragraph 10.6

(5) Final Engineering Report and Safety Analysis, Paragraph 9.2

(6) Final Engineering Report and Safety Analysis, Paragraph 10.2

(7) NIS Safety Review Report, April 1988

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

		Three Reactor Coolant Pumps Operating
*1.	Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip is not credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip is credited
2.	Pressurizer Pressure: High	≤ 2220 psig
3.	Nuclear Overpower	
	a. High Setting**	≤ 109% of indicated full power
	b. Low Setting	≤ 25% of indicated full power
***4.	Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341
***5.	Coolant Flow	≥ 85% of indicated full loop flow

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* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function.

** The nuclear overpower trip high setting is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 5% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 5%.

***May be bypassed at power levels below 10% of full power.

3.5 INSTRUMENTATION AND CONTROL

3.5.1 REACTOR TRIP SYSTEM INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.1-1.

OBJECTIVE: To delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

SPECIFICAION: As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.5.1-1 shall be OPERABLE.

ACTION: As shown in Table 3.5.1-1.

BASIS: During plant operations, the complete instrumentation systems will normally be in service.[1] Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits.[2] Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design.[1][3] This Standard outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

References:

(1) Final Engineering Report and Safety Analysis, Section 6.

(2) Final Engineering Report and Safety Analysis, Section 6.2.

(3) NIS Safety Review Report, April 1988

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTION UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 2	1 1	2 2	1, 2 3*, 4*, 5*	1 7
2. Power Range, Neutron Flux, Overpower Trip	4	2	3	1, 2	2#
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	4	1**	4	1, 2	28#
4. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
5. Source Range, Neutron Flux					
A. Startup	2	1**	2	2##	4
B. Shutdown	2	1**	2	3*, 4*, 5*	7
C. Shutdown	2	0	1	3, 4, and 5	5
6. NIS Coincidentor Logic	2	1	2	1, 2 3*, 4*, 5*	29 7
7. Pressurizer Variable Low Pressure	3	2	2	1####	6#
8. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
9. Pressurizer High Level	3	2	2	1	6#
10. Reactor Coolant Flow					
A. Single Loop (Above 50% of Full Power)	1/loop	1/loop in any operating loop	1/loop in each operating loop	1	6#
B. Two Loops (Below 50% of Full Power)	1/loop	1/loop in two operating loops	1/loop in each operating loop	1####	6#
11. Steam/Feedwater Flow Mismatch	3	2	2	1, 2	6#
12. Turbine Trip-Low Fluid Oil Pressure	3	2	2	1####	6#

TABLE 3.5.1-1 (Continued)

TABLE NOTATION

*	With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.
**	A "TRIP" will stop all rod withdrawal.
#	The provisions of Specification 3.0.4 are not applicable.
##	Below the Source Range High Voltage Cutoff Setpoint.
###	Below the P-7 (At Power Reactor Trip Defeat) Setpoint.
####	Above the P-7 (At Power Reactor Trip Defeat) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are met:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be returned to the untripped condition for up to 2 hours for surveillance testing of other channels per Specification 4.1.

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the Source Range High Voltage Cutoff Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the Source Range High Voltage Cutoff Setpoint.

b. Above the Source Range High Voltage Cutoff Setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

However, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

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

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.5.2 as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours.

ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 28 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirements, within one hour reduce THERMAL POWER such that T_{ave} is less than or equal to 551.5°F, and place the rod control system in the manual mode.

ACTION 29 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirements, be in at least HOT STANDBY within 6 hours; however, one channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.1, provided the other channel is OPERABLE.

TABLE 3.5.6-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
Pressurizer Water Level	3	2
Auxiliary Feedwater Flow Indication*	2/steam generator	1/steam generator
Reactor Coolant System Subcooling Margin Monitor	2	1
PORV Position Indicator (Limit Switch)	1/valve	1/valve
PORV Block Valve Position Indicator (Limit Switch)	1/valve	1/valve
Safety Valve Position Indicator (Limit Switch)	1/valve	1/valve
Containment Pressure (Wide Range)	2	1
Steam Generator Water Level (Narrow Range)	1/steam generator	1/steam generator
Refueling Water Storage Tank Level	1	1
Containment Sump Water Level (Narrow Range)**	2	1
Containment Water Level (Wide Range)	2	1
Neutron Flux (Wide Range)	2	1

* Auxiliary feedwater flow indication for each steam generator to provided by one channel of steam generator level (Wide Range) and one channel of auxiliary feedwater flow rate. These comprise the two channels of auxiliary feedwater flow indication for each steam generator.

** Operation may continue up to 30 days with one less than the total number of channels OPERABLE.

3.11 AXIAL OFFSET MONITORING

APPLICABILITY: MODE 1 above 90% RATED THERMAL POWER.

OBJECTIVE: To provide corrective action in the event that the axial offset monitoring system limits are approached.

SPECIFICATION: The incore axial offset limits shall not exceed the functional relationship defined by:

For positive offsets: $IAO = \frac{2.89/P - 2.1225}{0.03021} - 3.0$

For negative offsets: $IAO = \frac{2.89/P - 2.1181}{-.03068} + 3.0$

where

IAO = incore axial offset

P = fraction of rated thermal power

ACTION:

A. With IAO exceeding the limit defined by the specification, within 1 hour action shall be taken to reduce THERMAL POWER until IAO is within specified limits or such that THERMAL POWER is restricted to less than 90% of RATED THERMAL POWER.

B. With one or both excore axial offset channel(s) inoperable, as an alternate, one OPERABLE NIS channel for each inoperable excore axial offset channel, shall be logged every two hours to determine IAO.

C. With no method for determining IAO available, within 1 hour action shall be taken such that THERMAL POWER is reduced to less than 90% of RATED THERMAL POWER until a method of determining axial offset is restored.

BASIS: The percent full power axial offset limits are conservatively established considering the core design peaking factor, analytical determination of the relationship between core peaking factors and incore axial offset considering a wide range of maneuvers and core conditions, and actual measurements relating incore axial offset to the axial offset monitoring systems. The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to 3 percent on incore axial offset to allow for uncertainties in the correlation. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and incore axial offset has changed in a manner

warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of IAO shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10.

Reducing power in cases when limits are approached or exceeded, will assure that design limits which were set in consideration of accident analyses are not exceeded. In the event that no method exists for determining IAO, actions are specified to reduce THERMAL POWER to 90% of RATED THERMAL POWER. However, if axial offset channel(s) are inoperable, hand calculational methods of determining IAO from OPERABLE NIS channels can be employed until OPERABILITY of the axial offset channel(s) is restored.

References:

(1) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973

(2) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974.

(3) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Propriety Class 3.

4.1.1 OPERATIONAL SAFETY ITEMS

APPLICABILITY: Applies to surveillance requirements for items directly related to Safety Standards and Limiting Conditions for Operation.

OBJECTIVE: To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

SPECIFICATION:

A. Reactor Trip System instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.1.

B. Equipment and sampling tests shall be as specified in Table 4.1.2.

C. The specific activity and boron concentration of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.1.2., Item 1a.

D. The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1.2., Item 1b.

E. All control rods shall be determined to be above the rod insertion limits shown in Figure 3.5.2.1 by verifying that each analog detector indicates at least 21 steps above the rod insertion limits, to account for the instrument inaccuracies, at least once per shift during Startup conditions with K_{eff} equal to or greater than one.

F. The position of each rod shall be determined to be within the group demand limit and each rod position indicator shall be determined to be OPERABLE by verifying that the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) agree within 35 steps at least once per shift during Startup and Power Operation except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) at least once per 4 hours.

G. During MODE 1 or 2 operation each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

H. Instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.3.

TABLE 4.1.1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R	N.A.
2. Power Range, Neutron Flux	S	D (2,3) R (3,4)	M	N.A.	N.A.
3. Power Range, Neutron Flux, Dropped Rod Rod Stop	N.A.	N.A.	M	N.A.	N.A.
4. Intermediate Range, Neutron Flux	S	R (3,4)	S/U (1),M	N.A.	N.A.
5. Source Range, Neutron Flux	S	R (3)	S/U (1),M	N.A.	N.A.
6. NIS Coincidentor Logic	N.A.	N.A.	N.A.	N.A.	M (5)
7. Pressurizer Variable Low Pressure	S	R	M	N.A.	N.A.
8. Pressurizer Pressure	S	R	M	N.A.	N.A.
9. Pressurizer Level	S	R	M	N.A.	N.A.
10. Reactor Coolant Flow	S	R	Q	N.A.	N.A.
11. Steam/Feedwater Flow Mismatch	S	R	M	N.A.	N.A.
12. Turbine Trip-Low Fluid Oil Pressure	N.A.	N.A.	N.A.	S/U (1,6)	N.A.

TABLE 4.1.1 (Continued)

TABLE NOTATION

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER. Adjust channel if absolute difference greater than 2 percent.

(3) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) - The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(5) - Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(6) - Setpoint verification is not applicable.

TABLE 4.1.2
MINIMUM EQUIPMENT CHECK AND SAMPLING FREQUENCY

		Check	Frequency
1a. Reactor Coolant Samples	1.	Gross Activity Determination	At least once per 72 hours. Required during Modes 1, 2, 3 and 4.
	2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days. Required only during Mode 1.
	3.	Spectrascopic for $\bar{E}$ (1) Determination	1 per 6 months (2) Required only during Mode 1.
	4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours, (3) whenever the specific activity exceeds 1.0 μ Ci/gram DOSE EQUIVALENT I-131 or 100/ $\bar{E}$ (1) μ Ci/gram. b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.
	5.	Boron concentration	Twice/Week

(1) $\bar{E}$ is defined in Section 1.0.
(2) Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.
(3) Until the specific activity of the reactor coolant system is restored within its limits.

TABLE 4.1.3

MINIMUM FREQUENCIES FOR TESTING, CALIBRATING, AND/OR CHECKING OF INSTRUMENT CHANNELS

	Channels	Surveillance	Minimum Frequency
1.	Axial Offset	Calibration	At each refueling shutdown
		Check	Once per shift
2.	Reactor Coolant Temperature	Calibration	At each refueling shutdown
		Test	Once per month
		Check	Once per shift
3.	Pressurizer Pressure Input to Safety Injection Actuation	Calibration	At each refueling shutdown
		Test	Once per month
4.	Rod Position Recorder	Calibration	At each refueling shutdown
		Check, comparison with digital readouts	Once per shift during operation
5.	Charging Flow	Calibration	At each refueling shutdown
6.	Boric Acid Tank Level	Calibration	At each refueling shutdown
		Test	Once per month
7.	Residual Heat Pump Flow	Calibration	At each refueling shutdown
8.	Volume Control Tank Level	Calibration	At each refueling shutdown.
		Test	Once per month during MODES 1 and 2
9.	Hydrazine Tank Level	Calibration	At each refueling shutdown
		Test	One per month during operation

TABLE 4.1.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
Pressurizer Water Level	M	R
Auxiliary Feedwater Flow Indication*	M	R
Reactor Coolant System Subcooling Margin Monitor	M	R
PORV Position Indicator	M	R
PORV Block Valve Position Indicator	M	R
Safety Valve Position Indicator	M	R
Containment Pressure (Wide Range)	M	R
Steam Generator Water Level (Narrow Range)	M	R
Refueling Water Storage Tank Water Level	M	R
Containment Sump Water Level (Narrow Range)	M	R
Containment Water Level (Wide Range)	M	R
Neutron Flux (Wide Range)	M	R**

* See footnote of Table 3.5.6-1.

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

(4) The battery charger for 125 volt DC Bus No. 1 will supply at least 800 amps DC at 130 volts DC for at least 8 hours,

(5) The battery charger for 125 volt DC Bus No. 2 will supply at least 45 amps DC at 130 volts DC for at least 8 hours, and

(6) The battery charger for the UPS will supply at least 10 amps AC at 480 volts AC for at least 8 hours as measured at the output of the UPS inverter.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80%, 85% for Battery Bank No. 1, of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.4.D.2.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

E. The required Safety Injection System Load Sequencers shall be demonstrated OPERABLE at least once per 31 days on a staggered test basis, by simulating SISLOP* conditions and verifying that the resulting interval between each load group is within ± 10% of its design interval.

F. The required diesel generators and the Safety Injection System Load Sequencers shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

Attachment 1

EXISTING TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

CHANNEL CALIBRATION

1.2 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds with the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated. 83 11/2/84

CHANNEL CHECK

1.3 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL TEST

1.4 A CHANNEL TEST shall be the injection of a simulated signal into the channel to verify its proper response including, where applicable, alarm and/or trip initiating action.

CONTAINMENT INTEGRITY

1.5 CONTAINMENT INTEGRITY shall exist when:

(1) All non-automatic containment isolation valves (or blind flanges) are closed.

(2) The equipment door is properly closed.

(3) At least one door in each personnel air lock is properly closed.

(4) All automatic containment isolation valves are operable.

CORE ALTERATION

1.6 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position. 83 11/2/84

CORRELATION CHECK

1.7 A CORRELATION CHECK shall be an engineering analysis of an incore flux map wherein at least one point along the incore versus excore correlation data plot is obtained. 17 12/20/7

CORRELATION VERIFICATION

1.8 A CORRELATION VERIFICATION shall be the engineering analysis of incore flux maps wherein multiple points along the incore versus excore correlation data plot are obtained. 83 11/2/84 17 12/20/7

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." 83 11/2/84

FIRE SUPPRESSION WATER SYSTEM

1.10 A FIRE SUPPRESSION WATER SYSTEM shall consist of a water source(s), pump(s), and distribution piping with associated isolation valves (i.e., system header, hose standpipe and spray header isolation valves). 40 3/8/78

FREQUENCY NOTATION

1.11 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1. 83 11/2/84

GASEOUS RADWASTE TREATMENT SYSTEM

1.12 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

MEMBER(S) OF THE PUBLIC

1.13 MEMBER(S) OF THE PUBLIC shall include all individuals who by virtue of their occupational status have no formal association with the plant. This category shall include nonemployees of the licensee who are permitted to use portions of the site for recreational, occupational, or purposes not associated with plant functions. This category shall not include nonemployees such as vending machine servicemen or postmen who, as part of their formal job function, occasionally enter an area that is controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

OFFSITE DOSE CALCLUATION MANUAL (ODCM)

1.14 An OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

OPERABLE - OPERABILITY

1.15 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s). 83 11/2/84

OPERATIONAL MODE - MODE

1.16 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2. 61 6/11/81

PROCESS CONTROL PROGRAM

1.17 The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analysis, and formulation determination by which SOLIDIFICATION of radioactive wastes from liquid systems is assured.

PURGE-PURGING

1.18 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. 83 11/2/84

RATED THERMAL POWER

1.19 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 1347 Mwt.

REPORTABLE EVENT

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11/2/84

1.20 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

RESIDUAL HEAT REMOVAL (RHR) TRAIN

1.21 An RHR TRAIN shall be a train of components that includes: one RHR pump aligned with one RHR heat exchanger; one component cooling water pump aligned with the same RHR heat exchanger and with one component cooling water heat exchanger; and one salt water pump aligned with the same component cooling water heat exchanger.

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10/4/84

SHUTDOWN MARGIN

1.22 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

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11/2/84

SITE BOUNDARY

1.23 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

1.24 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.25 A SOURCE CHECK is the qualitative assessment of a channel response when the channel sensor is exposed to a radioactive source.

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3/5/87

STAGGERED TEST BASIS

1.26 A STAGGERED TEST BASIS shall consist of:

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11/2/84

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.27 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. 83 11/2/84

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.28 A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required setpoint within the required accuracy. 82 11/7/84

UNRESTRICTED AREA

1.29 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.30 A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atomspheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components. 83 11/2/84

VENTING

1.31 VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

E - AVERAGE DISINTEGRATION ENERGY

1.32 E is the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines and tritium with half lives greater than 15 minutes, making up at least 95% of the total non-iodine and non-tritium activity in the coolant. 96 3/5/87

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

APPLICABILITY: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

OBJECTIVE: To maintain the integrity of the reactor coolant system and to prevent the release of excessive amounts of fission product activity to the coolant.

SPECIFICATION: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded. 60 6/8/8

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1

TABLE 2.1

		Three Reactor Coolant Pumps Operating	
*1.	Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip is not credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip is credited	97 4/7/86
2.	Pressurizer Pressure: High	≤ 2220 psig	49 7/19/79
**3.	Nuclear Overpower	≤ 109% of indicated full power	
***4.	Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341	60 6/8/81
***5.	Coolant Flow	≥ 85% of indicated full loop flow	49 7/19/79

* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function. 97 4/7/86

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than 10% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above 10%.

***May be bypassed at power levels below 10% of full power.

Revised: 4/28/87
Typo Revised: 5/6/87

BASIS: Safety Limits

1. Reactor Coolant System Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the containment atmosphere. In addition, the failure of components of the Reactor Coolant System could result in damage to the fuel and pressurization of the containment. A safety limit of 2735 psig (110% of design pressure) has been established which represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section VIII.

2. Plant Operating Transients

In order to prevent any significant amount of fission products from being released from the fuel to the reactor coolant, it is necessary to prevent clad overheating both during normal operation and while undergoing system transients. Clad overheating and potential failure could occur if the heat transfer mechanism at the clad surface departs from nucleate boiling. System parameters which affect this departure from nucleate boiling (DNB) have been correlated with experimental data to provide a means of determining the probability of DNB occurrence. The ratio of the heat flux at which DNB is expected to occur for a given set of conditions to the actual heat flux experienced at a point is the DNB ratio and reflects the probability that DNB will actually occur.

It has been determined that under the most unfavorable conditions of power distribution expected during core lifetime and if a DNB ratio of 1.44 should exist, not more than 7 out of the total of 28,260 fuel rods would be expected to experience DNB. These conditions correspond to a reactor power of 125% of rated power. Thus, with the expected power distribution and peaking factors, no significant release of fission products to the reactor coolant system should occur at DNB ratios greater than 1.30.(1) The DNB ratio, although fundamental, is not an observable variable. For this reason, limits have been placed on reactor coolant temperature, flow, pressure, and power level, these being the observable process variables related to determination of the DNB ratio. The curves presented in Figure 2.1.1 represent loci of conditions at which a minimum DNB ratio of 1.30 or greater would occur. (1)(2)(3) 49 7/19/79

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less. 97 4/7/8

2. Variable Low Pressure, Loss of Flow, and Nuclear Overpower Trips

These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 10%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above.

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.(3)(6)

Revised 4/28/87
Typo Revised 5/6/87

References:

(1) Amendment No. 10 to the Final Engineering Report and Safety Analysis, Section 4, Question 3

(2) Final Engineering Report and Safety Analysis, Paragraph 3.3

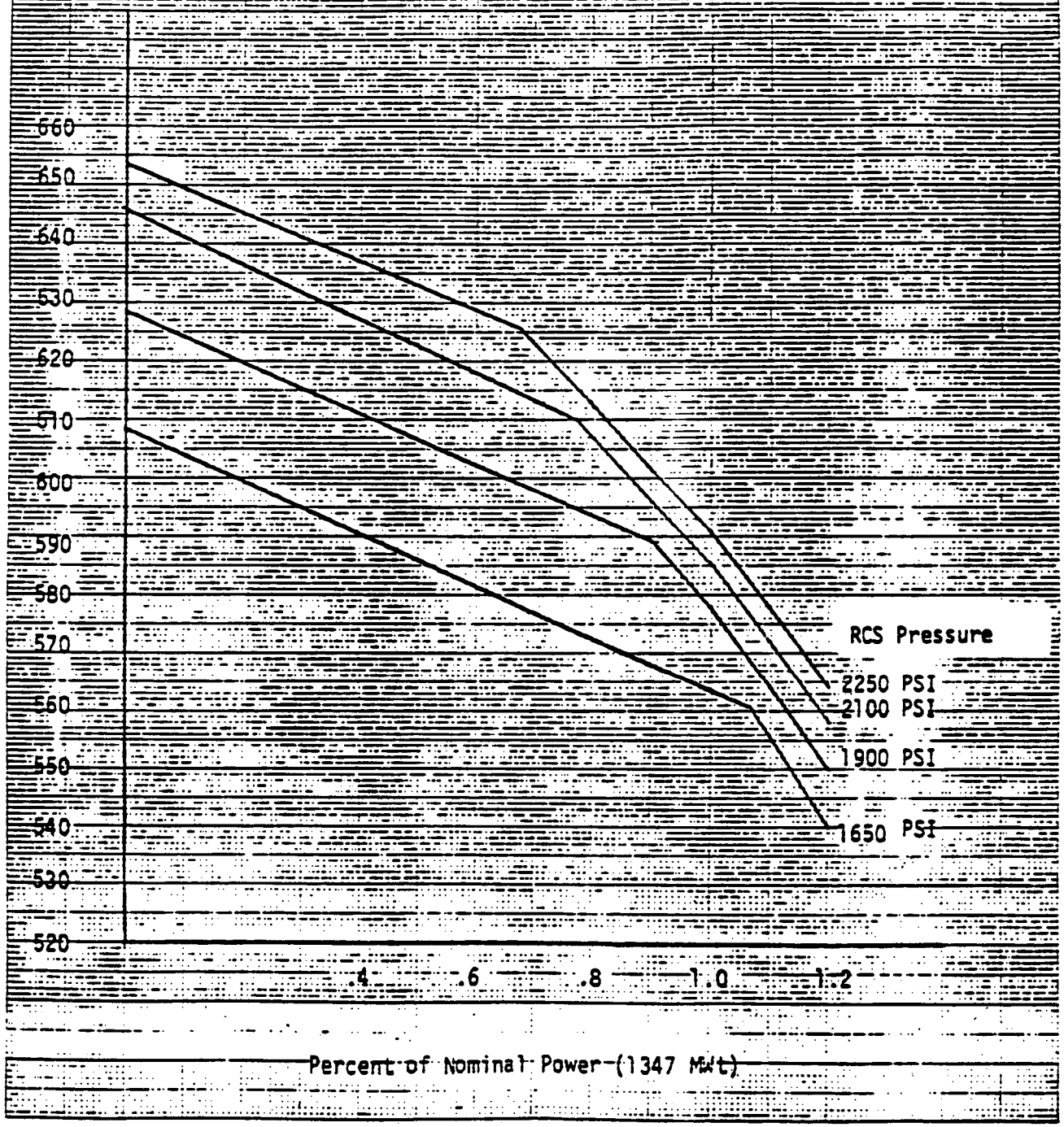
(3) Final Engineering Report and Safety Analysis, Paragraph 6.2

(4) Final Engineering Report and Safety Analysis, Paragraph 10.6

(5) Final Engineering Report and Safety Analysis, Paragraph 9.2

(6) Final Engineering Report and Safety Analysis, Paragraph 10.2

Figure 2.1.1
Safety Limits
Temperature, Power, Pressure
RCS Flow ≥ 195,000 GPM



3.5 INSTRUMENTATION AND CONTROL

3.5.1 REACTOR TRIP SYSTEM INSTRUMENTATION

APPLICABILITY: As shown in Table 3.5.1-1.

OBJECTIVE: To delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

SPECIFICATION: As a minimum, the reactor trip system instrumentation channels and interlocks of Table 3.5.1-1 shall be OPERABLE. 83 11/2/8

ACTION: As shown in Table 3.5.1-1.

BASIS: During plant operations, the complete instrumentation systems will normally be in service.[1] Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits.[2] Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design.[1] This Standard outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

References:

(1) Final Engineering Report and Safety Analysis, Section 6.

(2) Final Engineering Report and Safety Analysis, Section 6.2.

TABLE 3.5.1-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	7
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
4. Source Range, Neutron Flux					
A. Startup	2	1**	2	2##	4
B. Shutdown	2	1**	2	3*, 4*, 5*	7
C. Shutdown	2	0	1	3, 4, and 5	5
5. Pressurizer Variable Low Pressure	3	2	2	1####	6#
6. Pressurizer Fixed High Pressure	3	2	2	1, 2	6#
7. Pressurizer High Level	3	2	2	1	6#
8. Reactor Coolant Flow					
A. Single Loop (Above 50% of Full Power)	1/loop	1/loop in any operating loop	1/loop in each operating loop	1	6#
B. Two Loops (Below 50% of Full Power)	1/loop	1/loop in two operating loops	1/loop in each operating loop	1####	6#
9. Steam/Feedwater Flow Mismatch	3	2	2	1, 2	6#

TABLE 3.5.1-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
10. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1####	6#

TABLE 3.5.1-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal.

** A "TRIP" will stop all rod withdrawal.

\# The provisions of Specification 3.0.4 are not applicable.

\#\# Below the Source Range High Voltage Cutoff Setpoint.

\#\#\# Below the P-7 (At Power Reactor Trip's Active) Setpoint.

\#\#\#\# Above the P-7 (At Power Reactor Trip's Active) Setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 8 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.1.

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11/2/84

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the Source Range High Voltage Cutoff Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the Source Range High Voltage Cutoff Setpoint.

b. Above the Source Range High Voltage Cutoff Setpoint but below 10 percent of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10 percent of RATED THERMAL POWER.

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

ACTION 5 - With the number of OPERABLE channels one less than the Mininum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.5.2 as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 8 hours.

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ACTION 7 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

3.5.6 ACCIDENT MONITORING INSTRUMENTATION

APPLICABILITY: MODES 1, 2 and 3. 83 11/2/84

OBJECTIVE: To ensure reliability of the accident monitoring instrumentation. 64 12/16/81

SPECIFICATION: The accident monitoring instrumentation channels shown in Table 3.5.6-1 shall be OPERABLE.

ACTION:

A. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5.6-1, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

B. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.5.6-1, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours. 83 11/2/84

C. The provisions of Specification 3.0.4 are not applicable.

BASIS: The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations." 64 12/16/81

References:

(1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

(2) NRC letter dated November 1, 1983, from D. G. Eisenhut to all Pressurized Water Reactor Licensees, NUREG-0737 Technical Specification (Generic Letter No. 83-37).

TABLE 3.5.6-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
Pressurizer Water Level	3	2
Auxiliary Feedwater Flow Indication*	2/steam generator	1/steam generator
Reactor Coolant System Subcooling Margin Monitor	2	1
PORV Position Indicator (Limit Switch)	1/valve	1/valve
PORV Block Valve Position Indicator (Limit Switch)	1/valve	1/valve
Safety Valve Position Indicator (Limit Switch)	1/valve	1/valve
Containment Pressure (Wide Range)	2	1
Steam Generator Water Level (Narrow Range)	1/steam generator	1/steam generator
Refueling Water Storage Tank Level	1	1
Containment Sump Water Level (Narrow Range)**	2	1
Containment Water Level (Wide Range)	2	1

* Auxiliary feedwater flow indication for each steam generator is provided by one channel of steam generator level (Wide Range) and one channel of auxiliary feedwater flow rate. These comprise the two channels of auxiliary feedwater flow indication for each steam generator.

** Operation may continue up to 30 days with one less than the total number of channels OPERABLE.

Typo Revised: 11/28/84
Revised: 11/16/84
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11/2/84

3.11 CONTINUOUS POWER DISTRIBUTION MONITORING

Applicability: Applies to axial offset limit.

Objective: To provide corrective action in the event that the axial offset monitoring system limits are approached. 17 12/20/74

Specification:

A. The incore axial offset limits shall not exceed the functional relationship defined by: 21 5/13/75 17 12/20/74

For positive offsets: $IAO = \frac{2.89/P - 2.1225}{0.03021} - 3.0$

For negative offsets: $IAO = \frac{2.89/P - 2.1181}{-.03068} + 3.0$ 60 6/8/81

where

IAO = incore axial offset 17 12/20/74

P = fraction of rated thermal power 21 5/13/75

B. If the incore limit defined by Specification A, as measured by the excore axial offset system, is exceeded by both axial offset monitoring channels, reactor power shall be reduced until Specification A is satisfied. 18 1/10/75 -

C. If it is determined that one of the excore axial offset monitoring channels is inoperable, the other axial offset channel shall be used to provide power distribution information. In addition, one NIS channel current shall be logged every two hours and axial offset information determined from this data until the inoperable channel has been returned to service. 17 12/20/74

D. If both channels should be declared inoperable, at least three NIS channel currents shall be logged every two hours and offset information determined from these data. If no method for determining axial offset is available, reactor power shall be reduced to 90% of rated thermal power.

Basis: The percent full power axial offset limits are conservatively established considering the core design peaking factor, analytical determination of the relationship between core peaking factors and incore axial offset considering a wide range of maneuvers and core conditions, and actual measurements relating incore axial offset to the 21 5/13/75 17 12/20/74

axial offset monitoring systems. The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to 3 percent on incore axial offset to allow for uncertainties in the correlation. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and incore axial offset has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of Specification A shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10.

Reducing power in cases when limits are approached or exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. Prior to installation of the axial offset monitoring system, the NIS system was used to monitor axial offset and showed there is considerable margin between axial offset normally seen and limits established in consideration of design peaking factors. 17 12/20/74

References:

(1) Supporting Information on Periodic Axial Offset Monitoring, San Onofre Nuclear Generating Station, Unit 1, September, 1973 -

(2) Supporting Information on the Continuous Axial Offset Monitoring System, San Onofre Nuclear Generating Station, Unit 1, July, 1974.

(3) Description and Safety Analysis, Including Fuel Densification, San Onofre Nuclear Generating Station, Unit 1, Cycle 5, January, 1975, Westinghouse Non-Propriety Class 3. 21 5/13/75

4.1.1 OPERATIONAL SAFETY ITEMS

Applicability: Applies to surveillance requirements for items directly related to Safety Standards and Limiting Conditions for Operation.

Objective: To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

A. Instrumentation shall be checked, tested, and calibrated as indicated in Table 4.1.1.

B. Equipment and sampling tests shall be as specified in Table 4.1.2.

C. The specific activity and boron concentration of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.1.2., Item 1a. 38 12/20/77

D. The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1.2., Item 1b.

E. All control rods shall be determined to be above the rod insertion limits shown in Figure 3.5.2.1 by verifying that each analog detector indicates at least 21 steps above the rod insertion limits, to account for the instrument inaccuracies, at least once per shift during Startup conditions with K_{eff} equal to or greater than one.

F. The position of each rod shall be determined to be within the group demand limit and each rod position indicator shall be determined to be OPERABLE by verifying that the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) agree within 35 steps at least once per shift during Startup and Power Operation except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the rod position indication system (Analog Detection System) and the step counter indication system (Digital Detection System) at least once per 4 hours. 61 6/11/81

G. During MODE 1 or 2 operation each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days. 83 11/2/84

TABLE 4.1.1

MINIMUM FREQUENCIES FOR TESTING, CALIBRATING, AND/OR CHECKING OF INSTRUMENT CHANNELS

Channels	Surveillance	Minimum Frequency	
1. Power Range	Calibration	At each refueling shutdown	
	Calibration, using heat balance	Once per day during MODE 1	83 11/2/84
	Test	Once per month	
	Check	Once per shift	
2. Intermediate Range	Test of log level	Prior to each start-up if test has not been performed within 7 days	31 10/26/7
	Test of period circuits using known exponentially varying input currents	Prior to each start-up if test has not been performed within 7 days	
	Check log and period channels	Once per shift when in service	
3. Source Range	Test	Prior to each start-up if test has not been performed within 7 days	
	Check	Prior to each startup and then once per shift.	
4. Axial Offset	Calibration	At each refueling shutdown	
	Check	Once per shift during MODE 1	17 12/20/7
5. Reactor Coolant Temp.	Calibration	At each refueling shutdown	
	Test	Once per month	
	Check	Once per shift	

TABLE 4.1.1 (Continued)

Channels	Surveillance	Minimum Frequency	
6. Reactor Coolant Flow	Calibration	At each refueling shutdown	17 12/20/7
	Test	Every 3 months	12 9/17/73
	Check	Once per shift	
7. Pressurizer Pressure	Calibration	At each refueling shutdown	17 12/20/7
	Test	Once per month	
	Check	Once per shift	
8. Pressurizer Level	Calibration	At each refueling shutdown	17 12/20/7
	Test	Once per month	
	Check	Once per shift	
9. Variable low pressure calculator	Calibration	At each refueling shutdown	17 12/20/7
	Check	Once per shift	12 9/17/73
10. Rod position recorder	Calibration	At each refueling shutdown	17 12/20/7
	Check, comparison with digital readouts	Once per shift during operation	12 9/17/73

TABLE 4.1.1 (Continued)

Channels	Surveillance	Minimum Frequency	
11. Steam Generator Level, including Flow Mismatch	Calibrate, Level and Flow	At each refueling shutdown	17 12/20/7
	Test Flow Mismatch	Once per month during operation and prior to resumption of operation when a shutdown period extends the test interval beyond one month	12 9/17/73
	Check Level and Flow	Once per shift during MODES 1 and 2	
12. Charging Flow	Calibration	At each refueling shutdown	12 12/20/7
13. Boric Acid Tank Level	Calibration	At each refueling shutdown	17 12/20/7
	Test	Once per month	
14. Residual Heat Pump Flow	Calibration	At each refueling shutdown	17 12/20/7
15. Volume Control Tank Level	Calibration	At each refueling shutdown.	17 12/20/7
	Test	Once per month during MODES 1 and 2	
16. Hydrazine Tank level	Calibration	At each refueling shutdown	
	Test	One per month during operation	
17. Reactor Trip on Turbine Trip	Calibration	At each refueling shutdown.	
	Test	Test prior to each startup if test has not been performed within 7 days.	83 11/02/8

TABLE 4.1.2
MINIMUM EQUIPMENT CHECK AND SAMPLING FREQUENCY

		Check	Frequency	
1a. Reactor Coolant Samples	1.	Gross Activity Determination	At least once per 72 hours. Required during Modes 1, 2, 3 and 4.	96 3/5/87
	2.	Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days. Required only during Mode 1.	
	3.	Spectrascopic for $\bar{E}$ (1) Determination	1 per 6 months (2) Required only during Mode 1.	74 12/6/84
	4.	Isotopic Analysis for Iodine Including I-131, I-133, and I-135.	a) Once per 4 hours, (3) whenever the specific activity exceeds 1.0 μ Ci/gram DOSE EQUIVALENT I-131 or 100/ $\bar{E}$ (1) μ Ci/gram.	
			b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	38 12/20/7
	5.	Boron concentration	Twice/Week	

(1) $\bar{E}$ is defined in Section 1.32.
(2) Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer. 96 3/5/87
(3) Until the specific activity of the reactor coolant system is restored within its limits.

TABLE 4.1.2 (continued)

		Check	Frequency	
1.b	Secondary Coolant Samples	1. Gross Activity Determination	At least once per 72 hours. Required only during Modes 1, 2, 3 and 4.	
		2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. Required only during Modes 1, 2, 3 and 4.	74 12/6/83
			b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit. Required only during Modes 1, 2, 3, and 4.	

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Typo Revision: 1/23/84

			Check	Frequency	
2.	Safety Injection Water Samples	a.	Boron Concentration	Monthly when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test beyond 1 month	12 9/17/73
3.	Control Rod Drop	a.	Verify that all rods move from full out to full in, in less than 2.7 seconds	At each refueling shutdown	
4.	(Deleted)				61 6/11/81
5.	Pressurizer Safety Valves	a.	Pressure Setpoint	At each refueling shutdown	
6.	Main Steam Safety Valves	a.	Pressure Setpoint	At each refueling shutdown	
7.	Main Steam Power Operated Relief Valves	a.	Test for Operability	At each refueling shutdown	
8.	Trisodium Phosphate Additive	a.	Check for system availability as delineated in Technical Specification 4.2	At each refueling shutdown	34 4/1/77
9.	Hydrazine Tank Water Samples	a.	Hydrazine concentration	Once every six months when the reactor is critical and prior to return of criticality when a period of subcriticality extends the test interval beyond six months	
10.	Transfer Switch No. 7	a.	Verify that the fuse block for breaker 8-1181 to MCC 1 is removed	Monthly, when the reactor is critical and prior to returning reactor to critical when period of subcriticality extended the test interval beyond one month	

TABLE 4.1.2 (Continued)

			Check		Frequency	
11.	MOV-LCV-1100 C Transfer Switch	a.	Verify that the fuse block for either breaker 8-1198 to MCC 1 or breaker 42-12A76 to MCC 2A is removed.	Same as Item 10 above		
12.	Emergency Siren Transfer Switch	a.	Verify that the fuse block for either breaker 8-1145 to MCC 1 or breaker 8-1293A to MCC 2 is removed	Same as Item 10 above		34 4/1/77
13.	Communication Power Panel Transfer Switch	a.	Verify that the fuse block for either breaker 8-1195 to MCC 1 or breaker 8-1293B to MCC 2 is removed	Same as Item 10 above		
14.a.	Spent Fuel Pool Water Level		Verify water level per Technical Specification 3.8	a.	Once every seven days when spent fuel is being stored in the pool.	
b.	Refueling Pool Water Level			b.	Within two hours prior to start of and at least once per 24 hours thereafter during movement of fuel assemblies or RCC's.	43 9/25/78
15.	Reactor Coolant Loops/ Residual Heat Removal Loops	a.	Per Technical Specifications 3.1.2.c and 3.1.2.D, in Mode 1 and Mode 2 verify that all required reactor coolant loops are in operation and circulating reactor coolant.	a.	Once per 12 hours	80 10/4/84
		b.	Per Technical Specification 3.1.2.E, in Mode 3 verify			

TABLE 4.1.2 (Continued)

Check	Frequency	
1. At least two required reactor coolant pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days	
2. The steam generators associated with the two required reactor coolant pumps are operable with secondary side water level $\geq$ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours	
3. At least one reactor coolant loop is in operation and circulating reactor coolant.	3. Once per 12 hours	
c. Per Technical Specification 3.1.2.F, in Mode 4 verify		80 10/4/84
1. At least two required (RC or RHR) pumps are operable with correct breaker alignments and indicated power availability.	1. Once per 7 days	
2. The required steam generators are operable with secondary side water level $\geq$ 256 inches of narrow range on cold calibrated scale.	2. Once per 12 hours	
3. At least one reactor coolant loop/RHR train is in operation and circulating reactor coolant.	3. Once per 12 hours	
d. Per Technical Specifications 3.1.2.G and 3.1.2.H, in Mode 5 verify, as applicable:		

TABLE 4.1.2 (Continued)

	Check	Frequency	
	1. At least one RHR train is in operation and circulating reactor coolant.	1. Once per 12 hours	
	2. When required, one additional RHR train is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days	
	3. When required, the secondary side water level of at least two steam generators is $\geq$ 256 inches of narrow range on cold calibrated scale.	3. Once per 12 hours	
e.	Per Technical Specification 3.8.A.3, in Mode 6, with water level in refueling pool greater than elevation 40 feet 3 inches, verify that at least one method of decay heat removal is in operation and circulating reactor coolant at a flow rate of at least 400 gpm.	e. Once per 12 hours	80 10/4/84
f.	Per Technical Specification 3.8.A.4, in Mode 6, with water level in refueling pool less than elevation 40 feet 3 inches, verify		
	1. At least one decay heat removal method is in operation and circulating reactor coolant.	1. Once per 12 hours	
	2. One additional decay heat removal method is operable with correct pump breaker alignments and indicated power availability.	2. Once per 7 days	

BASIS:

Calibration

Calibration should be performed at every reasonable opportunity in order to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels should be calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of approximately one year.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once-per-year (approximately) for the process system channels is considered acceptable.

Testing

The minimum testing frequency for those instrument channels connected to the safety system is based on an assumed "unsafe failure" rate of one per channel every four years. This assumption is, in turn, based on operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal.

The failure rate of one per channel every four years and the testing interval of two weeks imply that, on the average, each channel will be inoperable for 1.75 days per year, or 1.75/365 year. Since two channels must fail in order to negate the safety function, the probability of simultaneous failure of two channels (assuming only two to be in service) is 1.75/365 squared, or 2.3×10^{-5}. From this it can be inferred that in a three channel system the probability of simultaneous

failure of two channels is approximately 6.9×10^{-5}. This represents the fraction of time in which each three channel system would have one operable and two inoperable channels, and equals $6.9 \times 10^{-5} \times 8760$ hours per year, or (approximately) 36 minutes/year.

It must also be noted that to thoroughly and correctly test a channel, the channel components must be made to respond in the same manner and to the same type of input as they would be expected to respond to during their normal operation. This, of necessity, requires that during the test the channel be made inoperable for a short period of time. This factor must be, and has been, taken into consideration in determining testing frequencies.

Because of their greater degree of redundancy, the 1/3 and 2/4 logic arrays provide an even greater measure of protection and are thereby acceptable for the same testing interval. Those items specified for monthly testing are associated with process components where other means of verification provide additional assurance that the channel is operable, thereby requiring less frequent testing.

During a 2-year testing period, the Reactor Coolant Flow Trips for each loop were tested 40 times. In all the tests the trips operated precisely on set point. Also, during this period, there were no ´unsafe failures´ as defined above in the Reactor Coolant Flow Trips or any similar trip circuitry. All of these channels represent more than 30 years of service without a single ´unsafe failure´. Because of the demonstrated reliability of these instrument channels and particularly the Reactor Coolant Flow Trip, the testing interval of the Reactor Coolant Flow Trip has been extended to 3 months.

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication, etc. can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, the minimum checking frequency of once per shift is deemed adequate.

4.1.5 ACCIDENT MONITORING INSTRUMENTATION

APPLICABILITY: MODES 1, 2 and 3. 83 11/2/84

OBJECTIVES: To ensure the reliability of the accident monitoring instrumentation shown in Table 4.1.5-1.

SPECIFICATION: Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.1.5-1. 64 12/16/8

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

References: (1) NRC letter dated July 2, 1980, from D. G. Eisenhut to all pressurized water reactor licensees.

TABLE 4.1.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
Pressurizer Water Level	M	R
Auxiliary Feedwater Flow Indication*	M	R
Reactor Coolant System Subcooling Margin Monitor	M	R
PORV Position Indicator	M	R
PORV Block Valve Position Indicator	M	R
Safety Valve Position Indicator	M	R
Containment Pressure (Wide Range)	M	R
Steam Generator Water Level (Narrow Range)	M	R
Refueling Water Storage Tank Water Level	M	R
Containment Sump Water Level (Narrow Range)	M	R
Containment Water Level (Wide Range)	M	R

* See footnote of Table 3.5.6-1.

4.4 EMERGENCY POWER SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Emergency Power System. 82 11/7/84

OBJECTIVE: To verify that the Emergency Power System will respond promptly and properly when required.

SPECIFICATION:

A. The required offsite circuits shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignments and power availability. 84 11/14/84

B. The required diesel generators shall be demonstrated OPERABLE:

1. At least once per 31 days on a STAGGERED TEST BASIS by:

a. Verifying the diesel starts from standby conditions, 95 7/3/86

b. Verifying a fuel transfer pump can be started and transfers fuel from the storage system to the day tank, 34 4/1/77

c. Verifying the diesel generator is synchronized and running at 4500 kW ± 5% for ≥60 minutes, 95 7/3/86

d. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses,

e. Verifying the day tank contains a minimum of 290 gallons of fuel, and 34 4/1/77

f. Verifying the fuel storage tank contains a minimum of 37,500 gallons of fuel.

2. At least once per 3 months by verifying that a sample of diesel fuel from the required fuel storage tanks is within the acceptable limits as specified by the supplier when checked for viscosity, water and sediment. 84 11/14/84

C. AC Distribution

1. The required buses specified in Technical Specification 3.7, Auxiliary Electrical Supply, shall be determined OPERABLE and energized from AC sources other than the diesel generators with tie breakers open between redundant buses at least once per 7 days by verifying correct breaker alignment and power availability. 34 4/1/77

D. The required DC power sources specified in Technical Specification 3.7 shall meet the following: 84 11/14/84

1. Each DC Bus train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and power availability. 34 4/1/77

2. Each 125 volt battery bank and charger shall be demonstrated OPERABLE:

   a. At least once per 7 days by verifying that:

      (1) The parameters in Table 4.4-1 meet the Category A limits, and

      (2) The total battery terminal voltage is greater than or equal to 129 volts on float charge.

   b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that: 84 11/14/84

      (1) The parameters in Table 4.4-1 meet the Category B limits,

      (2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

      (3) The average electrolyte temperature of ten connected cells is above $61^{\circ}F$ for battery banks associated with DC Bus No. 1 and DC Bus No. 2 and above $48^{\circ}F$ for the UPS battery bank.

   c. At least once per 18 months by verifying that:

      (1) The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

      (2) The cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion material, 34 4/1/77

      (3) The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, 84 11/14/84

(4) The battery charger for 125 volt DC Bus No. 1 will supply at least 800 amps DC at 130 volts DC for at least 8 hours,

(5) The battery charger for 125 volt DC Bus No. 2 will supply at least 45 amps DC at 130 volts DC for at least 8 hours, and

(6) The battery charger for the UPS will supply at least 10 amps AC at 480 volts AC for at least 8 hours as measured at the output of the UPS inverter.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.4.D.2.d. 84 11/14/84

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

E. The required Safety Injection System Load Sequencers shall be demonstrated OPERABLE at least once per 31 days on a staggered test basis, by simulating SISLOP* conditions and verifying that the resulting interval between each load group is within ± 10% of its design interval.

F. The required diesel generators and the Safety Injection System Load Sequencers shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Simulating SISLOP *, and:

   a. Verifying operation of circuitry which locks out non-critical equipment, — 84 11/14/8

   b. Verifying the diesel starts from standby condition on the auto-start signal, energizes the emergency buses with permanently connected loads and the auto connected emergency loads** through the load sequencer (with the exception of the feedwater, safety injection, charging and refueling water pumps whose respective breakers may be racked-out to the test position) and operates for ≥5 minutes while its generator is loaded with the emergency loads, — 95 7/3/86; 84 11/14/8

   c. Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed. — 95 7/3/86

3. Verifying the generator capability to reject a load of 2611 kW without tripping. — 84 11/14/84

* SISLOP is the signal generated by coincident loss of offsite power (loss of voltage on Buses 1C and 2C) and demand for safety injection.

** The sum of all loads on the engine shall not exceed 4500 kW + 5%. — 95 7/3/86

TABLE 4.4-1

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable value for each connected cell (3)
Electrolyte Level	>Minimum level indication mark, and ≤1/4" above maximum level indication mark	>Minimum level indication mark, and ≤1/4" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥2.13 volts	≥2.13 volts (c)	>2.07 volts
Specific Gravity (a)	≥1.200(b)	≥1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells >1.205	Average of all connected cells ≥1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps when on charge.
(c) Corrected for average electrolyte temperature in accordance with IEEE STD 450-1980.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameter(s) are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

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11/14/84

Basis:

The normal plant Emergency Power System is normally in continuous operation, and periodically tested.[1]

The tests specified above will be completed without any preliminary preparation or repairs which might influence the results of the test.

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4/1/77

The tests will demonstrate that components which are not normally required will respond properly when required.

The surveillance requirements for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensure the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.4-1 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .020 below normal full charge specific gravity or a battery charger current that has stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below normal full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below normal full charge specific gravity, ensures the OPERABILITY and capability of the battery.

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11/14/84

Operating with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.4-1 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below normal full charge specific gravity, ensures that the decrease in rating will be

less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below normal full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

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11/14/84

Reference:

(1) Supplement No. 1 to Final Engineering Report and Safety Analysis, Section 3, Questions 6 and 8.