ATTACHMENT 1

EXISTING TECHNICAL SPECIFICATION

APPENDIX A

TECHNICAL SPECIFICATIONS

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SHUTDOWN MARGIN

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are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.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

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

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 an equal subintervals,

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

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

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

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.

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 order to meet acceptance criteria for certain secondary side transients, the pressurizer high level trip must be set at 50% narrow range level or less.(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, 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)

3. <u>Steam/Feedwater Flow Mismatch</u>

A significant mismatch of steam flow and feedwater flow to the steam generators occurs at greater than 50% power in the event of LONF and FLB. In the event of these transients, the 2 out of 3 mismatch trip logic will result in reactor trip on the order of 1 second after the initiating event. The safety analysis conservatively assumed that reactor trips would occur at 5 seconds and 10 seconds for LONF and FLB, respectively. The high and low settings assure that regardless of the type of mismatch occurring for individual loops, a protective reactor trip is provided, which satisfy the single failure criterion for the postulated events.(8)

4. <u>Reactor Coolant Pump Breaker Open</u>

The Reactor Coolant Pump (RCP) Breaker Open reactor trip provides a redundant trip to the low flow trip. The overcurrent trip of the RCP breakers protects the core following a locked rotor and the undercurrent trip of the RCP breakers protects the core following a sheared shaft. The trip settings are selected to meet the analysis assumptions that rods begin to drop 6.1 seconds after the initiating event. The Reactor Protection System Permissives change the trip on RCP breaker open to 2/3 loops instead of 1/3 loops at power levels below 50%.

## TABLE 2.1

## MAXIMUM SAFETY SYSTEM SETTINGS

| | Three Reactor Coolant Pumps Operating |
|---|---|
| 1. Pressurizer High Level | ≤ 50% Pressurizer Narrow Range Level |
| 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 |
| ***6. Steam/Feedwater Flow Mismatch | |
| a. Low+ Setting: | $\frac{\text{Steam Flow - Feedwater Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$ |
| b. High+ Setting: | $\frac{\text{Feedwater Flow - Steam Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$ |
| **7. Reactor Coolant Pump Breaker Open | |
| a. Overcurrent | ≤ 2400 amps |
| b. Undercurrent | ≥ 110 amps |
| c. Undervoltage | ≥ 60% of rated bus voltage |

* The nuclear overpower trip 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.

***May be bypassed at power levels below 50% of full power.

\+ High and Low feedwater flow relative to steam flow.

BASIS:

Specification 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and

## 3.1.3 COMBINED HEATUP, COOLDOWN AND PRESSURE LIMITATIONS

APPLICABILITY: Applies to heatup and cooldown of the reactor coolant system.

OBJECTIVE: To maintain the structural integrity of the reactor coolant system throughout the lifetime of the plant.

SPECIFICATION:

A. Reactor pressure and heatup and cooldown of the reactor coolant system during the first 16 years of equivalent full power operation shall be limited in accordance with Figures 3.1.3a and 3.1.3b. Thereafter, limits shall be based on neutron exposure equivalent to not less than 16 years of full power operation, and Figures 3.1.3a and 3.1.3b shall be updated accordingly (by formal license amendment application).*

B. Figures 3.1.3a and 3.1.3b shall be updated in accordance with the following criteria and procedures:

(1) The methods of Appendix G, "Protection Against Nonductile Failure", to Section III of the ASME Boiler and Pressure Vessel Code shall be used to obtain the allowable pressure-temperature relationships for the reactor coolant system.

(2) The curves in Figure 3.1.3c shall be used in predicting the reference nil-ductility temperature increase, $RT_{NDT}$ unless measurements on the irradiation specimens show $RT_{NDT}$s greater than those predicted by the curves, in which case a new curve having the same slope as the original shall be constructed.

C. The pressurizer heatup rate of 100°F/hour and cooldown rate of 200°F/hour shall not be exceeded.

D. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line as shown in Figures 3.1.3a.

BASIS: The initial Reference Nil Ductility Temperature ($RT_{NDT}$) for all reactor vessel material based on Charpy V-notch data, drop weight tests, and conservative estimates** is 82°F or less. The $RT_{NDT}$ at the 1/4 thickness location (location of Appendix G reference flaw tip) increases as a function of cumulative neutron exposure up to approximately 240°F for the core region of the reactor vessel after 30 years of operation.

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* Technical Specification 3.20.A(1) should be reevaluated for continued applicability of the low pressure PORV overpressure setpoint at any time the heatup and cooldown curves are changed.

** NRC Standard Review Plan Branch Technical Position MTEB 5-2.

A sixteen (16) equivalent full power year service period was chosen for the operational limits given in this specification because at the end of this period the limiting $RT_{NDT}$ of the reactor vessel at the 1/4 thickness location is approximately 217°F in the core region. This $RT_{NDT}$ is at least 50°F above the $RT_{NDT}$ of all other regions in the primary reactor coolant system.

The highest $RT_{NDT}$ of the core region material is determined by adding the radiation induced $\Delta RT_{NDT}$ for the applicable time period to the original $RT_{NDT}$ shown in the Table 3.1.3.1. The fast neutron ($E > 1Mev$) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1.3d. Using the applicable fluence at the end of the year period and the copper content of the material in question, the RTNDT is obtained from Figure 3.1.3c.

Values of $\Delta RT_{NDT}$ may continue to be determined in this manner unless measurements on the irradiation specimens show $\Delta RT_{NDT}$s greater than those predicted by the curves for the equivalent capsule exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from non-mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and discussed in detail in Reference 1.

The results of these calculations are provided in Reference 2.

The design heatup and cooldown rates for the pressurizer are 100°F/hour and 200°F/hour, respectively.

The vertical line portion of the criticality limit given in Figures 3.1.3a is at the minimum permissible temperature for the 2485 psig in-service hydrostatic test as required by Appendix G to 10CFR Part 50. The non-vertical portion of the criticality limit is shifted 40°F to the right of the heatup curve as required by Appendix G to 10CFR Part 50.

REFERENCES:

(1) "Pressure Temperature Limits" Section 5.3.2 of Standard Review Plan, NUREG-751087, 1975.

(2) S. E. Yanichko, et al, "Analysis of Capsule F from the Southern California Edison Company San Onofre Reactor Vessel Radiation Surveillance Program", WCAP 9520, May 1979.

(2) Containment Spray System

a. Two refueling water pumps are OPERABLE.

b. Two hydrazine additive pumps are OPERABLE.

c. Hydrazine tank level and hydrazine concentration comply with Specification 3.3.4.

(3) Valves and interlocks associated with each of the above systems are OPERABLE.

(4) Effective leakage from the recirculation loop outside the containment shall be less than 625 cc/hr as calculated from the following formula.

Effective Leakage = $a_1 \times L_1 + a_2 \times L_2 + a_3 \times L_3$

where,

$L_1$ = pump and valve leakage which drains to auxiliary building sump

$L_2$ = valve leakage in auxiliary building or doghouse

$L_3$ = valve leakage outside

$a_1$ = iodine release factor for leakage in auxiliary building sump

$a_2$ = iodine release factor for leakage in auxiliary building or doghouse

$a_3$ = iodine release factor for leakage outside the auxiliary building or doghouse

If effective leakage from the recirculating loop outside the containment exceeds 625 cc/hr, make necessary repairs to limit leakage to 625 cc/hr. within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

B. During critical operation or when the reactor coolant system temperature is above 200°F, as appropriate per Item A above, maintenance shall be allowed on any one of the following items at any one time:

(1) One motor-operated valve at a time (MOV 1100B or 1100D) in the recirculation loop upstream of the charging pump suction header for a period of time not longer than 72 consecutive hours.

(2) One refueling water pump and/or its associated discharge valve at a time, for a period not longer than 72 consecutive hours.

(3) One hydrazine pump and/or its associated discharge valve (SV600 or 601) at a time, for a period of time not longer than 72 consecutive hours.

(4) One charging pump for a period of time not longer than 72 consecutive hours.

(5) One of the two required component cooling water pumps for a period of time not longer than 72 consecutive hours.

(6) One of the two saltwater cooling pumps with the auxiliary saltwater cooling pump or screen wash pumps available as backup for a period of time not longer than 72 consecutive hours. The backup pump(s) shall be demonstrated operable by test within 1 hour of declaring the saltwater pump inoperable.

(7) One train of ESF switchover automatic trip for a period of time not to exceed 72 consecutive hours.

(8) One motor-operated valve at a time (MOV-1100C or MOV-1100E) in the VCT outlet line to the charging pump suction for a period of time not longer than 72 consecutive hours.

C. Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to demonstrate availability.

D. In the event of a failure of a recirculating pump, plant operation may continue provided operability of the remaining pump and its associated motive and control power are satisfactorily demonstrated on a daily basis, including verification that the containment spray bypass valves (CV517 and 518) are closed.

BASIS: The requirements of Specification A assure that before the reactor can be made critical, or before the reactor coolant system heatup is initiated, adequate engineered safeguards are OPERABLE. The limit of 625 cc/hr for the recirculation loop leakage ensures that the combined 0-2 hr EAB thyroid dose due to recirculating loop leakage and containment leakage will not exceed the limits of 10 CFR 100. The formula for determining the leakage incorporates consideration of the significance of leakage in different plant areas. The iodine release factor adjusts actual pump or valve leakage to account for the fraction of the iodine in the leakage which would actually be released to the atmosphere. The iodine release factors in the auxiliary building sump, the auxiliary building or doghouse, and outside are 0.05, 0.5, and 1.0, respectively.

When the reactor is critical or the reactor coolant system temperature is above 200°F, maintenance is allowed per Specifications B and C providing requirements in Specification C are met which assure OPERABILITY of the redundant component. The specified maintenance times are a maximum, and maintenance work will proceed with diligence to return the equipment to an operable condition as promptly as possible. OPERABILITY of the specified components shall be based on the results of Specification No. 4.2.

The allowable maintenance periods are based upon the repair of certain specific items. Based on the demonstration that equipment redundant to that removed from service is OPERABLE, it is reasonable to maintain the reactor at power over this short period of time.

In the unlikely event that the need for safety injection should occur:

-- functioning of one train will protect the core.[1][2][3][4] Containment sprays alone, however, will maintain containment pressure under design pressure.[5]

-- functioning of one of the two hydrazine additive pumps and associated discharge valve will effect introduction of hydrazine into containment spray water. This provides for absorption of airborne fission products and reduction of the thyroid doses associated with the maximum hypothetical accident to within 10 CFR 100 limits.

-- dissolution of 5400 pounds of anhydrous trisodium phosphate stored in the sump will ensure that the pH of the water in the sump will be greater than 7 within four (4) hours, so as to prevent chloride stress corrosion cracking of systems and components exposed to the circulating sump water.

In the event of inoperability of a recirculation pump, plant operation may continue since either pump is sufficient and a daily OPERABILITY demonstration of the remaining pump and its associated motive and control power provides assurance that it will be OPERABLE if required.

The switchover from injection to recirculation modes is a two part process, which consists of the automatic termination of the flow from SI/FW pumps including automatic pump trip and automatic closures of MOV's 850 A, B and C followed by manual realignment to recirculation from the containment sump. The automatic trip setpoint is bounded by the minimum water level in the sump to support recirculation for long term post-LOCA cooling and the minimum RWST level to support charging and containment spray during the manual realignment. The setpoint analysis conservatively determined the automatic trip setpoint to be 20% of the RWST level. The automatic trip setpoint is the result of the combination of the worst single active failure considering SIS and SISLOP conditions.

3.3.2 SHUTDOWN STATUS

APPLICABILITY: Applies to piping connections between the feedwater condensate system and the reactor coolant system.

OBJECTIVE: To preclude injection of feedwater condensate into the reactor coolant system when the reactor is shut down and to preclude the potential for overpressurization when water solid.

SPECIFICATION: A. When reactor fuel assemblies are in the vessel and the reactor coolant pressure is less than 500 psig, two "positive barriers" shall be provided between the feedwater condensate system and the piping connections to the reactor coolant system. Additionally, when the reactor coolant system is water solid at less than 500 psig, two positive barriers shall be provided between the safety injection system and piping connections to the reactor coolant system. A "positive barrier" is defined as follows:

(1) Motor Operated Valves

When closed and tagged with supply breakers open, except that power may be restored during no-flow tests of the safety injection system (Specification No. 4.2).

(2) Pneumatic/Hydraulic Operated Valves

When closed and the condition tagged with the respective hydraulic block valve closed except that they may be opened during no-flow tests of the safety injection system (Specification No. 4.2).

(3) Manually Operated Valves

When closed and condition tagged.

(4) Feedwater Pump (Overpressurization Protection Only)

When shutdown with the breaker in the racked out condition.

BASIS: Under normal conditions, system operational interlocks assure that injection of feedwater condensation to the reactor by

3.3.4 MINIMUM SOLUTION VOLUME HYDRAZINE CONCENTRATION IN THE HYDRAZINE TANK

APPLICABILITY: Applies to the inventory of spray additive solution.

OBJECTIVE: To insure availability of containment spray additive solution of required quality.

SPECIFICATION: When the reactor coolant system temperature is above 200°F, the hydrazine tank shall contain not less than 150 gallons of aqueous solution having a concentration of not less than 21 wt% $N_2H_4$.

BASIS: The hydrazine tank serves the purpose of acting as a reservoir of aqueous hydrazine solution for post-accident iodine removal.

100 gallons of $N_2H_4$ solution are required to reduce airborne iodine concentration in the event of a loss of coolant accident. By adding a 50% margin to this figure to enshire that NPSH to the spray addition pumps is maintained at all times, a total of 150 gallons is required. This amount fulfills requirements for safety injection operations.

Two auxiliary feedwater pumps, one steam driven and one electric driven, together with the steam system relief valves, provide core decay heat removal capability in the event of a sustained loss of off-site power. The electric driven pump is capable of being powered from the diesel. Either auxiliary feedwater pump has the capability to satisfy decay heat removal requirements from the core.(1)

The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions (including cooldown) for 32 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.


REFERENCES: (1) Supplement No. 1 to the Final Engineering Report and Safety Analysis, Section 3, Question 6.

3.4.2 MAXIMUM SECONDARY COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum radioiodine activity in the secondary coolant of the steam generators any time the primary coolant system temperature exceeds 200°F.

OBJECTIVE: To limit the consequences of an accidental release of secondary coolant to the environment.

SPECIFICATION: A. The specific activity of radioiodine in the secondary coolant shall be limited to 0.1 µCi/gm DOSE EQUIVALENT I-131.

ACTION: B. With the specific activity of the secondary coolant in excess of 0.1 µCi/gm DOSE EQUIVALENT I-131, the reactor shall be placed in cold shutdown within 36 hours.

BASIS: The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The restriction of 0.1 µCi/gram DOSE EQUIVALENT I-131 in the secondary system limits the 2 hour thyroid exposure dose to well within the guidelines of 10 CFR Part 100 at the site boundary under these accident conditions. This thyroid dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analysis.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

3.4.3 AUXILIARY FEEDWATER SYSTEM

APPLICABILITY: Applies to the auxiliary feedwater pumps and valves for MODEs 1, 2, and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat from the core.

SPECIFICATION: Two trains of auxiliary feedwater, including associated pumps and valves, shall be OPERABLE.

ACTION:

A. With one Train of auxiliary feedwater inoperable, restore the inoperable train to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

B. With both Trains of auxiliary feedwater inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

Two auxiliary feedwater trains and the steam system relief valves provide core decay heat removal capability in the event of a sustained loss of offsite power. Either auxiliary feedwater train has the capability to satisfy decay heat removal requirements from the core, with a delivered flow of at least 185 gpm per train with three intact main feedwater lines and pressurized steam generators, 100 gpm per train with two intact main feedwater lines and pressurized steam generators, and 175 gpm per train with two intact main feedwater lines and depressurized steam generators.

AFW System Train A pumps and valves consist of AFW pumps G-10S and G-10 and associated valves, including flow control valves FCV-2300A, FCV-2300B, and FCV-2300C.

AFW System Train B pump and valves consist of AFW pump G-10W and associated valves, including flow control valves FCV-3300A, FCV-3300B, and FCV-3300C.

3.4.4 AUXILIARY FEEDWATER STORAGE TANK

APPLICABILITY: Applies to the auxiliary feedwater storage tank for MODES 1, 2 and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat.

SPECIFICATION: A. The auxiliary feedwater storage tank (AFST) shall be OPERABLE with a usable water volume of at least 190,000 gallons of water.

ACTION: B. With the AFST inoperable, within 4 hours restore the AFST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions (including cooldown) for 32 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. In addition, the water volume will provide sufficient margin to account for spillage that occurs during a main feedwater line break with loss of AFW flow indication prior to isolation of the broken line. Spillage is assumed to last no longer than one hour until the broken loop is identified via RCS Loop Delta-T positive indication that will be evident for the two intact steam generators. The usable water volume limit is specified relative to the bottom of the tank indicated level range (i.e., level tap). The contained water volume below this datum provides a significant margin to the NPSH and vortexing limits above the highest AFW pump suction inlet in the tank, but is not considered available for purposes of this specification.

## 3.5 INSTRUCTIONS 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 with the SETPOINTS and RESPONSE TIMES as shown in Tables 3.5.1-2 and 3.5.1-3, respectively.

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 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTION UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| 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##### | 6# |
| 12. Turbine Trip-Low Fluid Oil Pressure | 3 | 2 | 2 | 1#### | 6# |
| 13. Reactor Coolant Pump Breaker Position | | | | | |
| 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# |
| 14. 4kV Bus 1C and Bus 2C Undervoltage | 2/bus | 1/bus from both buses | 1/bus from both buses | 1,2,3*,4* | 1# |

2) The SHUTDOWN MARGIN BASIS of Specification 3.5.2 is determined at least once per 12 hours.

3) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

4) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

5) The remainder of the rods in the group with the inoperable rod are aligned to within ±35 steps of the inoperable rod within one hour while maintaining the rod insertion limits of Figure 3.5.2.

<u>BASIS</u>: The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential affects of rod misalignment on associated accident analyses.

The misalignment allowance of Specification B, assures core performance within allowed design margins including allowance for the inaccuracy of the position signals.

## 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* | | |
| o Auxiliary Feedwater Flow Rate | 1/steam generator | 1/steam generator |
| o Steam Generator Water Level (Wide Range) | 1/steam generator | 1/steam generator |
| o Reactor Coolant System Loop Delta-T Indication | 1/loop | 1/loop |
| 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 |
| Refueling Water Storage Tank Level | 2 | 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 is provided by one channel of auxiliary feedwater flow rate (Train B), one channel of environmentally qualified steam generator wide range level (Train A), and one channel of RCS Loop Delta-T indication. These comprise the three types of indication of auxiliary feedwater flow for each steam generator.

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

## TABLE 3.5.7-1

## AUXILIARY FEEDWATER INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| a. Manual Actuation | 2 | 1 | 2 | 1, 2, 3 | 12 |
| b. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3 | 13 |
| c. Steam Generator Water Level-Low | | | | | |
| i. Train A | 3 | 2 | 2 | 1, 2, 3 | 14, 15 |
| ii. Train B | 3 | 2 | 2 | 1, 2, 3 | 14, 15 |
| d. AFW Train Interlocks* | | | | | |
| i. Low Flow Train B/ Start Train A Fow | | | | | |
| 1) Start Pump G10S/Open Pump G10 Discharge Valve CV-2620, AND | 2 | 1 | 2 | 1, 2, 3 | 35, 36 |
| 2) Start Pump G10/Open Pump G10S Discharge Valve MOV-1202 | 2 | 1 | 2 | 1, 2, 3 | 35, 36 |
| ii. Normal Flow Train B/ Stop Train A Flow | | | | | |
| 1) Stop Pump G10S/Close Pump G10 Discharge Valve CV-2620, OR | 2 | 2** | 2 | 1, 2, 3 | 35, 36 |
| 2) Stop Pump G10/Close Pump G10S Discharge Valve MOV-1202 | 2 | 2** | 2 | 1, 2, 3 | 35, 36 |

* A total of 4 flow switches monitor Train B flow and each switch represents a channel which provides the specified signals to Train A.

** Only 1 of 2 Channels is required to trip of 1 Channel has been disconnected per the requirements of ACTION 35.

## TABLE 3.5.8.1

## RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|
| 1. Gross Radioactive Monitors Providing Automatic Termination of Release | | |
| a. Liquid Radwaste Effluent Line (R-1218) | (1) | 16 |
| b. Steam Generator Blowdown [a] Effluent Line (R-1216) | (1) | 17 |
| c. Turbine Building Sumps Effluent Line (Reheater Pit Sump) (R-2100) | (1) | 18 |
| d. Yard Sump (R-2101) | (1) | 18 |
| e. Component Cooling Water System [b] (R-1217) | (1) | 19 |
| 2. Flow Rate Measurement Devices | | |
| a. Liquid Radwaste Effluent Line (FE-16, FE-18) | (1) | 20 |
| b. Circulating Water Outfall* | | |
| c. Steam Generator Blowdown Effluent* Line | | |

* Pump status, valve turns or calculations are utilized to estimate flow.

(a) Secondary coolant samples and activity analysis performed in accordance with T.S. 4.1, Table 4.1.2.

(b) Closed loop system. Monitor closes vent valve to isolate surge tank.

TABLE 3.5.8.1
(Continued)

TABLE NOTATION

ACTION 16 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

1. At least two separate samples which can be taken by a single person are analyzed in accordance with Specification 4.5.1.A., and;

2. At least two technically qualified persons verify the release rate calculations and discharge valving.

ACTION 17 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least $10^{-7}$ microcurie/ml;

1. At least once per 12 hours when the specific activity of the secondary coolant is > 0.01 mCi/gram DOSE EQUIVALENT I-131.

2. At least once per 24 hours when the specific activity of the secondary coolant is ≤ 0.01 mCi/gram DOSE EQUIVALENT I-131.

ACTION 18 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least $10^{-7}$ microcurie/ml.

ACTION 19 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, determine if there is leakage from the Component Cooling Water System to the Salt Water Cooling System. If leakage exists sample the Component Cooling Water System to estimate the activity being released via the Salt Water Cooling System at least once per 24 hours for gross activity (beta or gamma) at a lower limit of detection of at least $10^{-7}$ microcurie/ml.

ACTION 20 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in-situ may be used to estimate flow.

# TABLE 3.5.10-1

## RADIATION MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM SETPOINT | MEASUREMENT RANGE | ACTION |
|---|---|---|---|---|---|
| 1. AREA MONITORS | | | | | |
| a. Control Room Area (R-1231) | 1 | All | 1 mR/hr | $10^{2}$ -$10^{2}$mR/hr | 25 |
| b. Spent Fuel Pool Area (R-1236) | 1 | * | 25 mR/hr | $10^{2}$ -$10^{2}$mR/hr | 26 |
| c. Containment Radiation Monitor-High Range (R-1255, R-1257) | 2 | 1, 2, 3 & 4 | 10 R/hr | 1-$10^{8}$ R/hr | 27 |
| 2. PROCESS MONITORS | | | | | |
| a. Wide Range Gas Monitor (R-1254) | 1 | 1, 2, 3 & 4 | per ODCM | $10^{7}$-$10^{5}$ mCi/cc | 27 |
| b. Main Steam Dump and Safety Valve Channels (R-1256A&B, R-1258A&B) | 1/steamline | 1, 2, 3 & 4 | 1mR/hr (low)<br>1 R/hr (high) | $10^{1}$-$10^{4}$ mR/hr<br>$10^{1}$-$10^{4}$ R hr | 27 |

* With fuel in the spent fuel pool or building

| Δk/k | Event | Basis for Adequacy |
|---|---|---|
| 5% | Open reactor coolant | Provides adequate margin so that maintenance activities can be carried out with the reactor head removed.[1] |

Regarding internal pressure limitations, the containment design pressure of 46.4 psig would not be exceeded if the sphere internal pressure before a major loss of coolant accident was no greater than 3.4 psig. The design criteria also allows an internal vacuum not in excess of 2.0 psig. Thus, the specified limiting conditions for internal pressure are consistent with the design basis.[2] Although such design values could be exceeded without damage to the structure, it is considered that the importance of the containment function warrants the specified values.

Opening of the ventilation system backup valves, POV 9A and POV 10A, is not considered a violation of containment integrity during startup conditions provided that their corresponding in-line valves POV 9 and POV 10 are closed.

REFERENCES:

(1) Supplement No. 3 to Final Engineering Report and Safety Analysis, Question No. 2.

(2) Final Engineering Report and Safety Analysis, Paragraph 5.3.

3.6.2 CONTAINMENT ISOLATION VALVES

APPLICABILITY: MODES 1, 2, 3 and 4.

OBJECTIVE: To provide assurance that the containment isolation valves listed in Table 3.6.2-1 will function when initiated by appropriate sensors.

SPECIFICATION: The containment isolation valves specified in Table 3.6.2-1 shall be OPERABLE.

ACTION:

A. With one or more of the isolation valve(s) specified in Table 3.6.2-1 inoperable, for each affected penetration that is provided with two isolation valves and is open maintain at least one valve OPERABLE, and for all affected penetrations with either one or two isolation valves, one of the following Actions shall be taken:

1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

2. Isolate each affected penetration within 4 hours by use of at least one deactivated* power operated valve secured in the isolation position, or

3. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

4. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with Action A.2 or A.3 above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are taken.

BASIS: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

* Valve may be temporarily activated for valve position verification and testing. While the valve is activated by this note, Action A.1 shall be applied and any system(s) declared inoperable pursuant to Action B shall not be declared OPERABLE.

3.7 AUXILIARY ELECTRICAL SUPPLY

3.7.1 ELECTRICAL SUPPLY: OPERATING

APPLICABILITY: MODES 1, 2, 3, and 4

OBJECTIVE: To define those conditions of electrical power availability necessary to provide for safe reactor operation and to provide for the continuing availability of engineered safeguards.

SPECIFICATION:

a. One Southern California Edison Company and one San Diego Gas & Electric Company high voltage transmission line to the switchyard and two transmission circuits from the switchyard, one immediate and one delayed access, to the onsite safety-related distribution system shall be OPERABLE. This configuration constitutes the two required offsite circuits.

b. Two redundant and independent diesel generators shall be OPERABLE each with a total connected design load not to exceed 6,000 kW and with:

   1. A separate day tank containing a minimum of 290 gallons of fuel,

   2. A separate fuel storage system containing a minimum of 37,500 gallons of fuel, and

   3. A separate fuel transfer pump.

c. Train A Emergency AC Buses shall be OPERABLE, comprised of:

   1. 4,160 volt Bus 1C,
   2. 480 volt Buses 1 and 3, and associated station service transformers with tie breaker open.

d. Train B Emergency AC Buses shall be OPERABLE, comprised of:

   1. 4,160 volt Bus 2C,
   2. 480 volt Buses 2 and 4, and associated station service transformers with tie breaker open.

e. 120 volt AC Vital Buses 1, 2, 3, 3A, and 4 energized from associated inverters connected to DC Bus 1.

f. 120 volt AC Vital Buses 5 and 6 energized from associated inverters connected to DC Bus 2.

g. 125 volt DC Bus 1 shall be OPERABLE and energized from Battery No. 1, with at least one full capacity charger.

h. 125 volt DC Bus 2 shall be OPERABLE and energized from Battery No. 2, with at least one full capacity charger.

i. Two trains of Safeguards Load Sequencing Systems (SLSS) shall be OPERABLE.*

j. The MOV-850C Uninterruptible Power Supply (UPS) shall be OPERABLE and energized from the battery with its full capacity charger.**

k. Manual Transfer Switch 7 (MTS-7) shall be OPERABLE and energized from MCC-2.

l. Manual Transfer Switch 8 (MTS-8) shall be OPERABLE and energized from MCC-4.

ACTION:

A. With one of the required offsite circuits inoperable, demonstrate the operability of the remaining AC sources by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 8 hours thereafter and Surveillance Requirement B.1.a within 24 hours. Restore the circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. If one diesel generator is declared inoperable, demonstrate the operability of the two offsite transmission circuits and the remaining diesel generator by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 8 hours thereafter and Surveillance Requirement B.1.a within 24 hours. Restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. With one offsite circuit and one diesel generator of the above required AC electrical power sources inoperable, demonstrate the operability of the remaining AC sources by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 8 hours thereafter and Surveillance Requirement B.1.a within 8 hours. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Have at least two offsite circuits and two diesel generators OPERABLE within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

D. With one diesel generator inoperable as in B or C above, verify that: (1) all required systems, subsystems, trains,

*The automatic load function may be blocked in Mode 3 at a pressurizer pressure $\leq$ 1,900 psig.

**Applicable in MODES 1, 2, and 3 above 500 psig.

components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE; and (2) the steam-driven auxiliary feedwater pump is OPERABLE in MODES 1, 2, and 3. If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

E. With two required offsite circuits inoperable, demonstrate the operability of two diesel generators by performing Surveillance Requirement B.1.a of Technical Specification 4.4 within 8 hours, unless the diesel generators are already operating. Restore at least one of the inoperable sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 4 hours. Have at least two offsite circuits and two diesel generators OPERABLE within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

F. With two of the above required diesel generators inoperable, demonstrate the operability of two offsite circuits by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 2 hours thereafter. Restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

G. With less than the above trains of Emergency AC buses OPERABLE, restore the inoperable buses within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

H. With one AC Vital Bus either not energized from its associated inverter, or with the inverter not connected to its associated DC Bus: (1) re-energize the AC Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) re-energize the AC Vital Bus from its associated inverter connected to its associated DC bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

I. With one DC bus inoperable or not energized from its associated battery and at least one full capacity charger, re-energize the DC Bus from its associated battery and at least one full capacity charger within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

J. With one Safeguards Load Sequencing System inoperable, restore the inoperable sequencer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

K. With the MOV-850C UPS inoperable, or not energized from its associated battery and its full capacity charger, restore the UPS to OPERABLE status and re-energize the UPS from its associated battery and its full capacity charger within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

L. With MTS-7 inoperable or not energized from MCC-2, restore MTS-7 to OPERABLE status and re-energize MTS-7 from MCC-2 within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

M. With MTS-8 inoperable or not energized from MCC-4, restore MTS-8 to OPERABLE status and re-energize MTS-8 from MCC-4 within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

BASIS: The station is connected electrically to the Southern California Edison Company and San Diego Gas & Electric Company system via either of two physically independent high voltage transmission routes composed of four Southern California Edison Company high voltage lines and four San Diego Gas & Electric Company high voltage lines.

Of the four Southern California Edison Company lines, any one can serve as a source of power to the station auxiliaries at any time. Similarly, any of the four San Diego Gas & Electric Company lines can serve as a source of power to the station auxiliaries at any time. By specifying one transmission line from each of the two physically independent high voltage transmission routes, redundancy of sources of auxiliary power for an orderly shutdown is provided.

Similarly, either transformer A or B, along with transformer C, provide redundancy of 4,160 volt power to the auxiliary equipment, and in particular to the safety injection trains. In addition, each 4160 volt bus has an onsite diesel generator as backup.

In MODES 1, 2, 3 and 4, two diesel generators provide the necessary redundancy to protect against a failure of one of the diesel generator systems or in case one diesel generator system is taken out for maintenance, without requiring a reactor shutdown. This also eliminates the necessity for depending on one diesel generator to operate for extended periods without shutdown if it were required for post-accident conditions.

When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE. In addition, the ACTION STATEMENT requires a verification that the steam-driven auxiliary feedwater pump is OPERABLE in MODES 1, 2, and 3.

These requirements are intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component. The Safeguards Load Sequencing System is designed so that each sequencer starts and loads its associated diesel generator and sequences the ECCS loads upon receipt of a safety injection signal (SIS) and concurrent loss of voltage on its respective 4160 volt bus (i.e., upon a SISLOP).

The total connected design load on each diesel generator is restricted to 6,000 kW or less. This requirement was the result of a crankshaft crack propagation analysis (see Reference 1). The analysis postulated that the crankshaft initially has stress-induced surface cracks. The analysis then considered the effect of four types of diesel load histories on the growth of these cracks. Each load history consisted of repeated start-stop cycles with some steady state operation at full load (6,000 kW) between each start and its stop. The analysis concluded that for a crankshaft with a detectable size crack (10 mils deep), the number of start-stop cycles required to enlarge the crack until it becomes self-propagating (18 mils deep) under the full load steady state stresses represents the effective life of the crankshaft.

During normal operations, the 480 volt system is considered OPERABLE if the four 480 volt buses and four station service transformers are OPERABLE with respective tie breakers open. This will ensure that the 480V main breakers and transformers remain OPERABLE during the worst loading condition in case of a SIS without LOP.

The primary power source for Vital Buses 1, 2, 3, 3A, and 4 is Train A DC Bus 1. The alternate power source is available from MCC-2 through MTS-7. The 1987 RPS and ESF single failure analyses credited the Train B backup power to these vital buses through MTS-7.

Correct operation of the safety injection system is assured by the operability of the load sequencers and the UPS for MOV-850C and MOV-358 (MOV-850C UPS). Correct operation of the recirculation system is assured by the operability of the MOV-850C UPS which also supplies MOV-358.

Manual Transfer Switch 8 (MTS-8) provides the means to power MOV-883 and the MOV-850C UPS from either Train A or Train B. However, due to single failure considerations and environmental effects, MTS-8 is normally powered from MCC-4 on Train B. MOV-883 is the discharge valve from the RWST and must remain open during the safety injection phase and close with initiation of recirculation.

REFERENCE: (1) Report No FaAA-84-12-14 (Revision 1.0), Evaluation of Transient Conditions on Emergency Diesel Generator Crankshafts at San Onofre Nuclear Generating Station, Unit 1.

3.7.2 ELECTRICAL SUPPLY: SHUTDOWN

APPLICABILITY: MODES 5 and 6

OBJECTIVE: To define those conditions of available electrical power to ensure that the station can be maintained in the shutdown or refueling condition for extended periods.

SPECIFICATION:

a. One Southern California Edison Company or San Diego Gas and Electric Company high voltage transmission line to the switchyard and one transmission circuit from the switchyard, immediate or delayed access, to the onsite safety-related distribution system shall be OPERABLE.

b. One diesel generator shall be OPERABLE which is capable of automatic start, with a total connected design load not to exceed 6,000 kW and with:

   1. A day tank containing a minimum 290 gallons of fuel,

   2. A fuel storage system containing a minimum of 37,500 gallons of fuel, and

   3. A fuel transfer pump.

c. One train of AC buses shall be OPERABLE comprised of:

   1. 4,160 volt Bus 1C, and 480 volt Buses 1 and 3 with at least one associated station service transformer; OR

   2. 4,160 volt Bus 2C, and 480 volt Buses 2 and 4 with at least one associated station service transformer.

d. 120 volt Vital Buses 1, 2, and 4 energized from associated inverters connected to DC Bus 1.

e. One 125 volt DC Bus OPERABLE and energized from the associated battery with at least one full capacity charger.

ACTION: With less than the minimum required AC and DC electrical sources specified above, suspend all operations involving CORE ALTERATIONS or positive reactivity changes. Initiate corrective actions to energize the required electrical buses. Within 8 hours, depressurize and vent the RCS through at least a 1.75 square inch vent.

BASIS: In MODES 5 and 6, the requirement for one source of offsite power and one diesel generator to be OPERABLE will provide diverse and redundant electrical power sources in order that the station can be maintained in the COLD SHUTDOWN or REFUELING condition for extended time periods. Additionally, this

## 3.8 FUEL LOADING AND REFUELING

APPLICABILITY: Applies to fuel handling and refueling operations. For the applicable surveillance requirements, see Table 4.1.2.

OBJECTIVE: To prevent incidents during fuel handling operations that could affect public health and safety.

SPECIFICATIONS: A. During refueling operations (MODE 6):

1. Radiation levels in the containment and spent fuel building shall be monitored.

2. Core subcritical neutron flux shall be continuously monitored during the entire refueling period by not less than two neutron monitors, each with continuous visual indication and one with continuous audible indication.

3. For water levels in the refueling pool, greater than elevation 40 feet, 3 inches (See 7. below for reference evaluation), the following specifications shall apply:

   a. At least one of the following methods of decay heat removal shall be in operation and circulating reactor coolant at a flow rate of ≥ 400 gpm:

      (1) One RHR TRAIN.

      (2) One refueling water pump taking suction from the refueling pool through the recirculation heat exchanger (with supporting heat removal systems operating), and discharging via the safety injection system piping to one reactor coolant loop cold leg.

   b. With less than one method of decay heat removal in operation, except as provided in c. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system. Immediately initiate corrective action to return the required decay heat removal method to operating status as soon as possible. In addition, within four hours, close all containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

   c. The decay heat removal capability may be removed from operation for up to one hour per eight hour period.

6. The reactor shall be subcritical for at least 148 hours prior to movement of irradiated fuel in the reactor pressure vessel.

7. Borated water to insure the SHUTDOWN MARGIN as specified in Item A.(5) above shall be maintained to an elevation not less than 40 feet 3 inches in the refueling pool during movement of fuel assemblies and RCC's. Reference elevation is sea level, mean lower low water.

8. If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be carried out.

B. With fuel assemblies in the spent fuel storage pool:

1. Loads in excess of 1,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

2. Borated water to insure the SHUTDOWN MARGIN as specified in Item A(5) above shall be maintained to an elevation not less than 40 feet 3 inches in the spent fuel storage pool. Reference elevation is sea level, mean lower low water.

3. With the requirement of B(2) above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limits within four hours.

BASIS: During refueling the reactor cavity is filled with approximately 240,000 gallons of borated water whose concentration is sufficient to maintain the reactor subcritical by greater than 5% $\Delta k/k$ or to a boron concentration greater than or equal to 2,000 ppm, whichever is more restrictive. Operation of one method of decay heat removal is provided to assure continuous mixing flow of refueling water through the reactor vessel during the refueling period.(1) Borated water injection capability is provided as per Specification 3.2 Part A in the unlikely event there is any need during the refueling period.

## 3.9 MODERATOR TEMPERATURE COEFFICIENT (MTC)

APPLICABILITY: Applies to negative moderator temperature coefficient (MTC) during core operations whenever the nominal reactor coolant inlet temperature is greater than or equal to 528°F.

OBJECTIVE: To establish negative MTC limits for the core.

SPECIFICATION:

a. The MTC shall be less negative than $-3.8 \times 10^{-4}$ $\Delta k/k/°F$ for all rods withdrawn, end of cycle life (EOL) and the RATED THERMAL POWER condition.

b. In order to assure that the above negative MTC limit is not exceeded, the MTC shall be measured at any THERMAL POWER and compared to the predetermined, calculated negative MTC within 7 effective full power days (EFPD) of reaching an equilibrium boron concentration of 300 ppm. The predetermined calculated MTC value (at RATED THERMAL POWER conditions with all rods withdrawn) is determined as follows:

$-3.1 \times 10^{-4}$ $\Delta k/k/°F$ MTC at a nominal core inlet coolant temperature of 551.5°F, and MTC increasing linearly with decreasing inlet coolant temperature to $-2.5 \times 10^{-4}$ $\Delta k/k/°F$ at a nominal core inlet coolant temperature of 528.0°F.

ACTION: In the event this comparison indicates the MTC is more negative than the applicable value given above, the MTC shall be remeasured, and compared to the EOL MTC limit of $-3.8 \times 10^{-4}$ $\Delta k/k/°F$ at least once per 14 EFPD during the remainder of the cycle. If the measured MTC is more negative than the $-3.8 \times 10^{-4}$ $\Delta k/k/°F$ limit any time during the remainder of the cycle, the reactor shall be in HOT SHUTDOWN within 12 hours after exceeding the negative MTC limit.

BASIS: The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the San Onofre Unit 1 accident and transient analyses.

The limiting MTC used in the steam line break accident analysis is given as a function of $k_{eff}$ and average moderator temperature in Figure 14 of Amendment 18 to the FSAR. In order to ensure that the safety analysis remains valid. The reactor should not be operated with an MTC more negative than the limit implied by Figure 14 of Amendment 18.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

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 IAO to the axial offset monitoring systems(1). The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed IAO limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and IAO has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to functional relationship of the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION.

Reducing power until IAO is within the specified limits in cases when limits are exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to place the plant in MODE 2 within 6 hours. 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) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989

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

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

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

## 3.12 CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM

APPLICABILITY: Applies to the operational status of the control room emergency air treatment system.

OBJECTIVE: To identify those conditions of the control room emergency air treatment system which will ensure reliable and efficient operation, should the system be needed.

SPECIFICATION: Effective upon completion of field testing to the modified filter system.

A. Except as specified in Specification 3.12.B below, the control room emergency air treatment system shall be OPERABLE whenever the reactor is to be made or maintained critical. The system will be considered OPERABLE as long as the tests and analyses specified in Specification 4.11 are satisfactorily completed at the required intervals and the system is not removed from service.

B. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days.

C. If the conditions in 3.12.B cannot be met, reactor shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN condition with 24 hours.

BASIS: The control room emergency air treatment system is designed to filter the control room intake air during control room isolation. The system is placed in operation under administrative control when conditions warrant its use.

The system utilizes a fan, a high efficiency particulate absolute (HEPA) filter, pre-filters and a charcoal absorber bed. The pre-filters are installed before the charcoal bed to prevent clogging of the iodine adsorbers. The charcoal adsorbers reduce the potential intake of radioiodine to the control room.

The OPERABILITY requirements of this Specification in conjunction with the surveillance requirement of Specification 4.11 provide reasonable assurance that the system will operate, if needed, at a degree of efficiency equal to or better than that assumed in the Final Safety Analysis.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited time while repairs are made. If the system cannot be repaired within seven days, the reactor is shut down and brought to COLD SHUTDOWN within 24 hours.

## 3.13 SHOCK SUPPRESSORS (SNUBBERS) OPERABILITY

APPLICABILITY: Applies to safety related shock supperssors (snubbers).

OBJECTIVE: To define operability requirements of snubbers required to protect safety related piping from unrestricted motion when subjected to dynamic loading as might occur during a seismic event or severe transient.

SPECIFICATION:

A. During MODES 1, 2, 3, and 4 (MODES 5 and 6 for snubbers located in the systems required OPERABLE in those MODES) all snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

B. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.14.C on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

BASIS: Snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by in-service functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling outage intervals. Observed failures of these sample snubbers will require functional testing of additional units. Snubbers of rated capacity greater than 120,000 pounds are exempt from functional testing requirements because of the impracticability of testing such large units.

Hydraulic snubbers and mechanical snubber may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3.14 FIRE PROTECTION

3.14.1 FIRE SUPPRESSION WATER SYSTEM

APPLICABILITY: At all times.

SPECIFICATION: The fire suppression water system shall be OPERABLE with:

a. Three fire suppression pumps, each with a rated capacity of at least 1000 gpm, with their discharge aligned to the fire suppression header. One pump must be from Unit 1; the remaining two pumps may be selected from the four pumps available at San Onofre Units 1, 2, and 3.

b. Two separate water supplies (one from Unit 1 and one from Units 2 & 3), each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path from each required water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the first valve upstream of the water flow alarm device on each sprinkler or hose standpipe, and the first valve upstream of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.14.2, 3.14.3, and 3.14.5.

ACTION:

A. With one required pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

B. With the fire suppression water system otherwise inoperable, establish a backup fire suppression system within 24 hours.

BASIS: The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray, and/or sprinklers, Halon, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

The requirement for OPERABILITY of separate fire water supplies from Unit 1 and Units 2 and 3 (including supply, pumping capability, and piping) will assure two separate supply connections to the Unit 1 fire main loop in accordance with BTP 9.5-1, Appendix A (1976), Section E.2.c. In the event that portions of the fire suppression water systems are inoperable, backup fire fighting equipment are required to be made

TABLE 3.14.2.1

REQUIRED SPRINKLER AND SPRAY SYSTEMS

| Fire Area/ Zone | Hazard | Location | System Type |
|---|---|---|---|
| 1 | Reactor coolant pumps, RHR pumps, cable | Inner Containment Sphere | Deluge - borated water spray* |
| | Cable insulation outside secondary shield | Outer Containment Sphere | Deluge - borated water spray* |
| 2A | Charging Pumps | Charging Pump Room | Wet Pipe |
| 4B/4D | Cable Insulation | Cable Trays, Yard/Breezeway Area | Deluge water spray |
| 4D | Transformer oil | Station Service Transformer 1<br>Transformers 2 & 4 | Deluge water spray<br>Deluge water spray |
| 9A | Turbine lubricating oil and cable insulation | System #1 chemical treatment area | Deluge water spray |
| | Turbine lubricating oil and cable insulation | System #2 lube oil reservoir area (north half) | Deluge water spray |
| | Turbine lubricating oil and cable insulation | System #3 lube oil reservoir area (south half) | Deluge water spray |
| | Turbine lubricating oil | System #4 480V room wall & turbine building north wall | Wet pipe |
| | Turbine lubricating oil and cable insulation | System #5 north turbine building area protection | Wet pipe |
| | Hyrdogen seal oil | Hydrogen seal oil unit | Deluge water spray |
| 17A | Diesel Generator | North Diesel Generator | Pre-Action Sprinkler |
| 18 | Diesel Generator | South Diesel Generator | Pre-Action Sprinkler |

*This includes a refueling water pump, 240,000 gallons of borated water in the refueling water storage tank and associated system valves.

3.14.5 FIRE HOSE STATIONS

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

SPECIFICATION: The following fire hose stations shall be OPERABLE:

a. See Table 3.14.5.1

ACTION:

A. With one or more of the fire hose stations shown in Table 3.7-6 inoperable, route a fire hose* to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to plant workers, plant equipment, or the hose itself, a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose shall be mounted at the inoperable hose station.

B. The provision of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: In the event that a fire hose station is inoperable, the establishment of backup suppression in the affected areas is required to provide fire suppression capability until the inoperable system is restored to operability.

REFERENCES:

1. Fire Protection Program Review, BTP APCSB 9.5-1, San Onofre Nuclear Generating Station, Unit 1, March 1977; submitted to the NRC by letter dated March 16, 1977 in Docket No. 50-206.

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* Fire hose will be run within 1 hour of entering the ACTION statement if an operable water supply is not available within 250 feet of the area protected by the inoperable hose station, or two 150 feet hose packs (1-3/4") on the fire truck are not operable. Fire hose will be supplied by the fire department responding to a fire if an operable water supply is available within 250 feet of the area protected by the inoperable hose station. With the required hose station inside containment inoperable and containment integrity established, fire hose will be supplied only to the nearest access point.

TABLE 3.14.5.1

FIRE HOSE STATIONS

| Fire Area/ Zone | Location | Elevation | Hose Station Number |
|---|---|---|---|
| 1 | Inside Sphere | 42' - 0" | 25 |
| 2A | Reactor Auxiliary Building, Lower Level | 5' - 0" | 17 |
| 2P | Boric Acid Injection Pump Room | 20" - 0" | 16 |
| 4D | Turbine Plant Cooling Water Area | 14' - 0" | 8 |
| 9A | Chemical Feed and Lubrication-Oil Reservoir Area | 14" - 0" | 22 |
| 9A | East Feedwater Pump/Condenser Area | 8' - 6" | 5 |
| 9A | East Feedwater Pump/Condenser Area | 14' - 0" | 3 |
| 9A | East Feedwater Pump/Condenser Area | 14' - 0" | 4 |
| 9A | West Feedwater Pump/Condenser Area | 14' - 0" | 7 |
| 9A | West Feedwater Pump/Condenser Area | 8' - 6" | 6 |
| 9B | Turbine and Heater Decks | 35' - 6" | 10 |
| 9B | Turbine and Heater Decks | 35' - 6" | 11 |
| 9B | Turbine and Heater Decks | 42' - 0" | 12 |
| 9B | Turbine and Heater Decks | 42' - 0" | 13 |
| 9B | Turbine and Heater Decks | 35' - 6" | 14 |
| 9B | Turbine and Heater Decks | 35' - 6" | 15 |
| 12 | Administration/Control Building First Floor Single-Story Office Area | 20' - 0" | 26 |
| 11A | Administration/Control Building First Floor Health Physics and Locker Area | 20' - 0" | 27 |
| 2D | Cryogenic Building Back Yard Area | 20' - 0" | 24 |

## 3.14.6 FIRE DETECTION INSTRUMENTATION

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

SPECIFICATION: As a minimum, the following fire detection instrumentation shall be OPERABLE:

a. See Table 3.14.6.1

ACTION:

A. With one or more of the required fire detection instruments inoperable:

1. For areas* outside containment establish within 1 hour an hourly fire watch patrol.

2. For areas* inside containment, establish a fire watch patrol in containment, at least once per 8 hours or monitor the temperature at least once per hour at the locations listed below:

   a. Inside secondary shield:
      A minimum of 2 out of 3 for each RCP: RCP lower water bearing, RCP upper guide bearing, and RCP down thrust bearing;

      AND,

      A minimum of 2 out of 3 of: After RCP motor cooling fan unit, After RCP standby motor cooling fan unit, and RCP motor space

   b. Outside secondary shield:
      A minimum of 3 out of 5 of: Control rod cooler discharge, Control rod shroud air inlet, Reactor cavity air outlet, Sphere space, and Control rod cooler inlet

3. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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* Not required for areas that pose temporary radiation and/or life-threatening safety hazards. If the fire watch patrol cannot be restored within 24 hours, prepare and submit a Special Report to Commission pursuant to Specification 6.9.2 within the next 7 days outlining the action taken, the cause of hazards, and the plans and schedule for restoring the required fire watch/patrol.

TABLE 3.14.6.1

FIRE DETECTION INSTRUMENTS

| Fire Area/Zone | Location | Required Instruments Oparable Early Warning | Actuation |
|---|---|---|---|
| 1 | Containment Sphere | | |
| | Inside Secondary Shield | 8 | |
| | Outside Secondary Shield | 20 | |
| 2A | Reactor Auxiliary Bldg. Lower Level | 9 | |
| 4A&B | East and West Penetration Areas | 33 | |
| 4C | Doghouse | 2 | |
| 4B/4D | Cable Trays Yard/Breezeway Area | | 2 |
| 4D | Service Transformer 1 | | 2 |
| | Service Transformer 2 & 4 | | 2 |
| 4G | DSD Diesel Generator Enclosure | 6 | |
| | DSD Switchgear/Battery Room | 3 | |
| 7 | 480V Switchgear Room | | 8 |
| 8 | 4160V Switchgear Room | | 16 |
| 9A | Turbine Building Ground Floor Instrument Air Compressors | 1 | |
| | Exciter and MCC-4 Area | 16 | 2 |
| | Lube Oil Reservoir | 29 | 12** |
| 11A | Health Physics and Locker Room | 5* (4 at present) | |
| 11B | HVAC Equipment Room | 3 | |
| 12 | Offices 1st Floor Power Block | 7 | |

*Upon completion of DCP 3449.01.
**Includes 6 line-type detectors.

any fire having the potential of making the existing normal safe shutdown systems unavailable. The DSD incorporates:

Remote shutdown capability

Independent onsite power source

RCS charging capability

Auxiliary feedwater flow capability

Independently powered instrumentation and controls

Similarly, alternate shutdown methods rely on existing plant equipment used in off-normal modes to assure safe shutdown capability in the event of certain fires. Table 3.14.8.1 lists the minimum equipment required to be OPERABLE in order to provide these capabilities. Use of the equipment in Table 3.14.8.1, as it is used to mitigate certain fires, is described in the references.

In the event that one or more components listed in Table 3.14.8.1 is rendered inoperable for more than 7 days, the Technical Specifications permit continued plant operation for up to 60 days, if equivalent shutdown capability is provided. The equivalent shutdown capability provided when ASS/DSS equipment is inoperable depends on the specific equipment involved and, therefore, should be sufficient to assure that the intended shutdown actions can be accomplished, or that fires can be reasonably precluded during that time for which ASS/DSS equipment would otherwise be required, consistent with the ASS/DSS design basis. Temporary procedures or special fire watch patrols established to provide this equivalent capability should be approved by the Plant Manager prior to implementation.

Boron concentration at the PASS Skid will be measured utilizing either potentiometric analysis (auto-titrator) or PASS boronometer. Potentiometric analysis utilizing grab samples and an auto-titrator provides a very accurate and reliable method of determining boron concentration. This technique is utilized by Unit 1 chemists routinely in support of plant operations. Continuous sampling of boron concentration is not required since inadvertent dilution of the primary system as a result of the spurious actuation of equipment is not likely since design basis fires causing the loss of neutron source range monitors have been analyzed to not impact the operability of components which have the capability of supplying unborated water to the RCS make up path. The PASS boronometer remains available.

3.15 RADIOACTIVE LIQUID EFFLUENTS

3.15.1 LIQUID EFFLUENTS CONCENTRATION

APPLICABILITY: At all times.

OBJECTIVE: Maintain the concentration of radioactive liquid material released from the site below 10 CFR 20 limits.

SPECIFICATION:

A. The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2 x $10^{-4}$ μCi/ml.

B. ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

BASIS: This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within 1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3.16 RADIOACTIVE GASEOUS EFFLUENTS

3.16.1 DOSE RATE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose rate at the exclusion area boundary from radioactive gaseous effluents within 10 CFR 20 limits.

SPECIFICATION: A. The dose rate to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following values:

1. The dose rate limit for noble gases shall be ≤ 500 mrem/year to the total body and ≤ 3000 mrem/year to the skin, and

2. The dose rate limit for I-131, I-133, for tritium and for all radionuclides in particulate form with half lives greater than 8 days shall be ≤ 1500 mrem/year to any organ.

B. ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

BASIS: This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrating of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THF PUBLIC who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limit also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to ≤ 1500 mrem/year.

TABLE 3.18.1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

| Exposure Pathway and/or Sample | Number of Samples and Sample Locations[a] | Sampling and Collection Frequency[a] | Type and Frequency of Analyses |
|---|---|---|---|
| 1. AIRBORNE Radioiodine and Particulates | Samples from at least 5 locations<br>3 samples from offsite locations (in different sectors) of the highest calculated annual average ground level D/Q.<br><br>1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q.<br><br>1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction.[c] | Continuous operation of sampler with sample collection an required by dust loading but at least once per 7 days.[d] | Radioiodine cartridge. Analysis at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample When gross beta activity is ≥ 10 time the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days. |
| 2. DIRECT RADIATION[e] | At least 30 locations including an inner ring of stations in the general area of the SITE BOUNDARY and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations. | At least once per 92 days. | Gamma dose. At least once per 92 days. |

| | Exposure Pathway and/or Sample | Number of Samples and Sample Locations[a] | Sampling and Collection Frequency[a] | Type and Frequency of Analyses |
|---|---|---|---|---|
| 3. | WATERBORNE | | | |
| | a. Ocean | 4 Locations | At least once per month and composited quarterly | Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days. |
| | b. Drinking | 2 Locations | Monthly at each location. | Gamma isotopic and tritium analyses of each sample. |
| | c. Sediment | 4 Locations from Shoreline | At least once per 184 days. | Gamma isotopic analysis of each sample. |
| | d. Ocean | 5 Locations Bottom Sediments | At least once per 184 data. | Gamma isotopic analysis of each saeple. |
| 4. | INGESTION | | | |
| | a. Nonmigratory Marine Animals | 3 Locations | One sample from each group (listed below) will be collected in season, or at least once per 184 days it not seasonal. Groups to be sampled:<br>1. Fish-2 adult species such as flatfish, bass, parch or sheepshead.<br>2. Crustaceae-such as crab or lobster.<br>3. Mollunks-such as limpets, clams or seahares. | Gamma isotopic analysis an edible portions. |

## 3.19 SOLID RADIOACTIVE WASTE

APPLICABILITY: At all times.

OBJECTIVE: Ensure meeting the requirements for the SOLIDIFICATION and shipment of solid radwaste.

SPECIFICATION:

A. The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

B. ACTION:

1. With the provisions of the PROCESS CONTROL PROGRAM satisfied suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification/agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

## 3.20 OVERPRESSURE PROTECTION SYSTEMS

APPLICABILITY: Applies to operability of the overpressurization protection systems.

OBJECTIVE: To preclude the potential for exceeding 10 CFR 50, Appendix G, in the event of a pressure transient while water-solid.

SPECIFICATION: A. When the RCS pressure is ≤ 400 psig* and pressurizer water level is greater than 50%, at least one of the following overpressure protection systems shall be OPERABLE:

(1) Two power operated relief valves (PORVs) with a lift setting of ≤ 500 psig,** or

(2) A reactor coolant system vent(s) of ≥ 1.75 square inches.

ACTION: B. With one PORV inoperable when required in accordance with Specification A above, either restore the inoperable PORV to OPERABLE status within seven days or depressurize and vent the RCS through a 1.75 square inch vent(s) within the next eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

C. With both PORVs inoperable when required in accordance with Specification A above, depressurize and vent the RCS through at least a 1.75 square inch vent(s) within eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

D. In the event either the PORVs or the RCS vent(s) are used to mitigate a potential RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances indicating transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

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* The placing in service of the OMS at ≤ 400 psig is intended to assure that protection is provided whenever temperature is below 360°F. The alarm to arm the OMS being keyed to pressure assures that inadvertent opening of the PORVs does not occur due to placing the OMS into service with RCS pressure above the 500 psig initiation setpoint.

** The 500 psig setpoint is based on the current heatup and cooldown curves for 16 EFPY. The setpoint requires reevaluation for acceptability any time the curves are changed.

## 4.0 SURVEILLANCE REQUIREMENTS (GENERAL)

APPLICABILITY: Applies to the surveillance requirements to be implemented in these specifications.

OBJECTIVE: To define the conditions under which the surveillance requirements of Section 4 Specifications are applicable.

SPECIFICATION:

4.0.1 Surveillance Requirements shall be met during OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

BASIS:

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational

status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by

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)<br>R (3,4) | M | N.A. | N.A. |
| 3. Power Range, Neutron Flux, Dropped 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. |
| 13. Reactor Coolant Pump Breaker Position* | S | R | R | N.A. | N.A. |
| 14. 4kV Bus 1C and Bus 2C Voltage | N.A. | R | R | N.A. | N.A. |

*Applicable to Item 6 in Table 2.1

TABLE 4.1.2 (continued)

| | | Check | Frequency |
|---|---|---|---|
| 2. Safety Injection Line and RWST 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 |
| 3. Control Rod Drop | a. | Verify that all rods move from full out to full in, in less than 2.44 seconds | At each refueling shutdown |
| 4. (Deleted) | | | |
| 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 |
| 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. Not used. | | | |

| | Check | Frequency |
|---|---|---|
| 11. (Deleted) | | |
| 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. | Monthly, when the reactor is critical and prior to returning reactor to critical, when period of subcriticality extended the test interval beyond one month. |
| 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. | Monthly, when the reactor is critical and prior to returning reactor to critical, when period of subcriticality extended the test interval beyond one month. |
| 14a. 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. |
| 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 and in MODE 3 with reactor trip breakers closed, verify that all required reactor coolant loops are in operation and circulating reactor coolant. | a. Once per 12 hours. |
| | b. Per Technical Specification 3.1.2.E, in MODE 3 with the reactor trip breakers open, verify. | |

4.1.6 PRESSURIZER RELIEF VALVES

APPLICABILITY: Applies to the power operated relief valves (PORVs) an their associated block valves for MODES 1, 2 and 3.

OBJECTIVE: To ensure the reliability of the PORVs and block valves.

SPECIFICATION: A. Each PORV shall be demonstrated OPERABLE:

1. At least once per 31 days by performance of a CHANNEL TEST, which may include valve operation, and

2. At least once per 18 months by performance of a CHANNEL CALIBRATION.

B. Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel, unless the block valve is being maintained closed in order to meet the requirements of Specification 3.1.5.A.

C. The backup nitrogen supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by transferring motive power from the normal air supply to the nitrogen supply and operating the valves through a complete cycle of full travel.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the getting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

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

## 4.2 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM

### 4.2.1 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Safety Injection System and the Containment Spray System.

OBJECTIVE: To verify that the Safety Injection System and the Containment Spray System will respond promptly and properly if required.

SPECIFICATION: I. System Tests

A. Hot Safety Injection System Test

(1) When the plant is planned to be shutdown from MODE 1 operation and is planned to enter MODE 5 operation, a Hot SIS Test shall be performed in MODE 3 while RCS pressure is above 1500 psi but not more often than once every 9 months. The test shall include a determination of the force required to open valves HV 851 A and B and the margin of available actuation force.

(2) The test will be considered satisfactory if:

(a) control board indication and visual observations indicate all components have operated and sequenced properly. That is, the appropriate pumps have started and/or stopped and started, and all valves have completed their travel.

(b) the measured actuator force for both the HV-851 A and B valves is equal to or less than 10,000 $lb_f$.*

(3) If the measured actuator force of either HV-851 A or B is between 10,000 and 22,000 $lb_f$, the HV-851 A and B valves shall be considered OPERABLE but the future testing interval shall be accelerated as determined by the following equation:

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*Upon receipt of satisfactory data from continuing testing and analysis, the NRC staff will consider a request from Southern California Edison Company to change this number to more accurately reflect existing conditions.

$$T = T_L \frac{(22,000 - F)}{12,000}$$

where: $T$ = maximum time in days of operation allowed before next surveillance test is required

$T_L$ = time in days of operation since the last surveillance test

$F$ = measured actuator force

(4) If the measured actuator force of either HV-851 A or B is greater than 22,000 $lb_f$, test results shall be reported to the NRC pursuant to Specification 6.9.2 along with proposed corrective actions. NRC approval shall be obtained prior to returning the unit to service.

B. Trisodium Phosphate Test

(1) A test of the trisodium phosphate additive shall be conducted once every refueling to demonstrate the availability of the system. The test shall be performed in accordance with the following procedure:

(a) The three (3) storage racks are visually observed to have maintained their integrity.

(b) The three (3) racks, each with a storage capacity of 1800 pounds of anhydrous trisodium phosphate additive, are visually observed to be full.

(c) Trisodium phosphate from one of the sample storage racks inside containment shall be submerged without agitation, in 25±0.5 gallons of 150°F to 175°F distilled water borated to 3900±100 ppm boron.

(2) The test shall be considered satisfactory if the racks have maintained their integrity, the racks are visually observed to be full, and the trisodium phosphate dissolves to the extent that a minimum pH of 7.0 is reached within 4 hours of the start of the test.

B. Leakage Testing

(1) The recirculation loop outside containment (including the Containment Spray System) shall be pressurized at a pressure equal to or greater than the operating pressure under accident conditions at intervals not to exceed the normal plant refueling interval. Visual inspections for leakage shall be made and if leakage can be detected, measurements of such leakage shall be made. In addition, pumps and valves of the recirculation loop outside containment which are used during normal operation, shall be visually inspected for leakage at intervals not to exceed once every six months. If leakage can be detected, measurements of such leakage shall be made.

(2) The non-redundant Containment Spray System piping shall be visually inspected at intervals not to exceed the normal plant refueling interval. Observations made as part of compliance with Paragraph C, above, or Paragraph I.C(2) of Technical Specification 4.2 will be acceptable as visual inspection of portions of non-redundant Containment Spray System piping.

C. RWST Low Level Trips

Monthly, perform a CHANNEL TEST and every refueling interval, perform a CHANNEL CALIBRATION, of the SI/Feedwater Pump trip and the MOV 850A, 850B and 850C automatic closure on low-low Refueling Water Storage Tank level.

BASIS: The Safety Injection System is a principal plant safeguard. It provides means to insert negative reactivity and limits core damage in the event of a loss of coolant or steam break accident.

Preoperational performance tests of the components are performed in the manufacturer's shop. An initial system flow test demonstrates proper dynamic functioning of the system. Thereafter, periodic tests demonstrate that all components are functioning properly. For these tests, flow through the system is generally not required. However, in the case of the "Hot SIS Test," actual conditions of an SI event are simulated. This test is performed to assure that long-term set of the valve seat faces on HV-851 A and B has not caused the valves to become inoperable. The test is required to be performed as the plant is shutting down from MODE 1 in order to assure that the valves have not been disturbed (i.e., the long-term set is still in effect) and that full dynamic conditions that would occur during an actual SI event are simulated. When possible the test should be performed prior to stopping the feedwater pumps (this is not a requirement). This will further assure that the valves will be in the same condition as when required for an actual Safety Injection event since the discharge pressure of the feedwater pumps acting on the valves will keep them seated even considering any backpressure built up in the downstream SI header. Theequation used to determine future intervals if actuator force is between 10,000 lb. and 22,000 lb. is developed by shortening the interval in direct proportion to the degree to which the force exceeds 10,000 lb.. During the test, all components are verified to have operated and sequenced properly.

The tests required in this specification will demonstrate that all components which do not normally and routinely operate will operate properly and in sequence if required. The portion of the Recirculation system outside the containment sphere is effectively an extension of the boundary of the containment. The measurement of the recirculation loop leakage ensures that the calculated EAB 0-2 hr. thyroid dose does not exceed 10 CFR 100 limits.

The trisodium phosphate stored in storage racks located in the containment is provided to minimize the possibility of stress corrosion cracking of metal components during operation of the ECCS following a LOCA. The trisodium phosphate provides this protection by dissolving in the sump water and causing its final pH to be raised to 7.0 - 7.5. The requirement to dissolve trisodium phosphate from one of the sample storage racks in distilled water heated and borated, to the extent recirculating post LOCA sump water is projected to be heated and borated, provides assurance that the stored trisodium phosphate will dissolve as required following a LOCA. The sample storage racks are sized to contain 0.5 pounds of trisodium phosphate. Trisodium phosphate stored in the sample storage racks has a surface area to volume ratio of 1.33 whereas the trisodium phosphate stored in the main racks has a surface area to volume ratio of 1.15.

Visual inspection of the non-redundant piping in the Containment Spray System provides additional assurance of the integrity of that system.

Surveillance testing of the RWST low-low level main feedwater/safety injection pump trips and automatic closure of MOV 850A, 850B and 850C valves will ensure that these components will be available to complete their safety functions if required.

## 4.2.2 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES TESTING
(Surveillance Requirement)

APPLICABILITY: Applies to the operational status of the primary coolant system pressure isolation valves during MODES 1, 2 and 3.

OBJECTIVE: To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

SPECIFICATION: 1. Periodic leakage testing(a) on each valve listed in Table 3.3.5-1 shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

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(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. The minimum test differential pressure shall not be less than 150 psid.

B. Acceptance Criteria

Visual inspection for leakage shall be made and if leakage can be detected, measurements of such leakage shall be made. The maximum effective leakage shall be maintained in accordance with Section 3.3.1.A(4) of Appendix A Technical Specifications.

C. Test Schedule

Visual inspections of the recirculation loop outside containment (including the Containment Spray System)) shall be made at intervals not to exceed the normal plant refueling interval. In addition, pumps and valves of the recirculation loop outside containment which are used during normal operation, shall be visually inspected for leakage at intervals not to exceed once every six months.

V. Test Result Report

The results of Type A, B, and C leakage rate tests are submitted to the NRC in a summary technical report approximately three months after the conduct of the Type A tests. This report contains an analysis and interpretation of the Type A test results and a summary of periodic Type B and C tests performed since the last Type A test. Leakage rate test results from Type A tests that fail to meet the acceptance criteria specified in Section I.B above are reported in a separate attached summary report that includes an analysis of the test data, an instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to failure in meeting the acceptance criteria. Results and analysis of the supplemental verification test used to demonstrate the validity of the Type A test measurements are included.

VI. Containment Modification

Any major modification or replacement of a component that is part of the containment boundary is followed by Type A, B, or C tests as applicable. The results of such tests are included in the test result report described above and meet the respective acceptance criteria. Minor modifications or replacements performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

4.3.2 CONTAINMENT ISOLATION VALVES

APPLICABILITY: Applies to the containment isolation valves listed in Table 3.6.2-1 for MODES 1, 2, 3 and 4.

OBJECTIVE: To ensure reliability of containment isolation valves.

SPECIFICATION:

A. The isolation valves specified in Table 3.6.2-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test.

B. Each isolation valve specified in Table 3.6.2-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

1. Verifying that on containment isolation test signal, each automatic isolation valve actuates to its isolation position.

2. Verifying that on a containment radiation-high test signal, each purge supply and purge outlet automatic valve actuates to its isolation position.

BASIS: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

REFERENCES:

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

4.3.3 HYDROGEN MONITORS AND HYDROGEN RECOMBINERS

APPLICATION: MODES 1 and 2.

OBJECTIVE: To ensure reliability of the hydrogen monitors and hydrogen recombiners required for the detection control of hydrogen gas.

SPECIFICATION: A. Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

1. One volume percent hydrogen, balance nitrogen.

2. Four volume percent hydrogen, balance nitrogen.

B. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per year by verifying that a heater sheath temperature of at least 1225 ± 10°F can be attained.

C. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits or foreign materials, etc.), and

3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the test in Specification B above. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

BASIS: The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with radiolytic decomposition of water and corrosion of metals within containment.

4.4 EMERGENCY POWER SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Emergency Power System.

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.

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 performs a DG SLOW START[1] from standby conditions,

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

c. Verifying the diesel generator is synchronized and running at 6,000 kW (+100 kW, -500 kW) for ≥ 60 minutes,

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

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.

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[1]All diesel starts for testing and surveillance will be slow starts (greater than 24 seconds duration) except for the fast start required by Technical Specification 4.4.G conducted once per 18 months during shutdown and any other fast start required following specific maintenance involving the fast start capability.

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 without automatic SIS/SISLOP tripping circuitry open between redundant buses at least once per 7 days by verifying correct breaker alignment and power availability.

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

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.

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:

      (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 $150 \times 10^{-6}$ ohms, and

      (3) The average electrolyte temperature of ten connected cells is above 61°F for battery banks associated with DC Bus No. 1 and DC Bus No. 2 and above 48°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 anticorrosion material,

(3) The resistance of each cell-to-cell and terminal connection is less than or equal to $150 \times 10^{-6}$ ohms,

(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 invertor.

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 Safeguards Load Sequencing Systems (SLSS) 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 Safeguards Load Sequencing Systems (SLSS) shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

1. Simulating SISLOP*, and:

   a. Verifying operation of circuitry which locks out non-critical equipment,

   b. Verifying the diesel performs a DG FAST START 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,

   c. Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed.

2. Verifying the generator capability to reject a load of 4,000 kW without tripping. The generator voltage shall not exceed 4,800 volts and the generator speed shall not exceed 500 rpm (nominal speed plus 75% of the difference between nominal speed and the overspeed trip setpoint) during and following the load rejection.

G. Manual Transfer Switches

1. Verify once every 31 days that the fuse block for breaker 8-1181 in MCC-1 for MTS-7 is removed.

2. Verify once every 31 days that MTS-8 is energized from breaker 8-1480B from MCC-4 and the cabinet door is locked, and that breaker 8-1122 from MCC-1 is locked open.

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* SISLOP is the signal generated by a sequencer on coincident loss of voltage on its associated 4160 volt bus (Bus 1C or 2C) and demand for safety injection.

** The sum of all loads on the engine shall not exceed 6,000 kW.

H. Periodic maintenance, surveillance, overhaul and inspection of the required diesel generator shall comply with the following:

1. A diesel engine maintenance and surveillance program as described in the Safety Evaluation related to Amendment No. 123 to this Operating License will be implemented. Changes to this program will be subject to the provisions of 10 CFR 50.59.

2. The frequency of major diesel engine overhaul that is a part of the diesel engine maintenance and surveillance program shall be at least once every ten years. For this overhaul, one engine may be inspected during the refueling outage immediately prior to the ten years and the other engine inspected during the refueling outage immediately following the ten years. Alternatively, both inspections may be performed coincident with the 10-year reactor vessel inservice inspection. The 10-year overhaul interval shall be determined on a calendar basis from the date of completion of the last overhaul.

3. Oil hole locations in journals 8 through 12 on each crankshaft shall be inspected with liquid penetrant. This inspection shall be performed at each refueling outage or at the end of fifty[2] start-stop cycles on the engine since the previous inspection, whichever comes first. Indications found shall be evaluated with eddy current testing as appropriate.

   During each major engine overhaul, the fillets of main journal Nos. 4 through 12 should be inspected together with the oil holes, using liquid penetrant. Indications found shall be evaluated with eddy current testing as appropriate. In addition, these inspections should be performed for the oil holes and fillets in at least three of the crankpin journals at each major engine overhaul.

   If during the oil hole and fillet inspections described above, cracks are found in the oil holes or in other crankshaft surfaces, these findings are to be reported to the NRC within 24 hours. The affected engine is to be considered inoperable and is not to be restored to OPERABLE status until the disposition and/or corrective actions have been approved by the NRC staff.

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[2]Start-stop cycles associated with idle (no load) engine operation at 200 rpm or less need not be counted toward the limit of fifty.

4. Cylinder blocks shall be inspected for "ligament" cracks, "stud-to-stud" cracks and "stud-to-end" cracks as defined in the report[3] by Failure Analysis Associates, Inc. (FaAA) entitled "Design Review of TDI R-4 and RV-4 Series Emergency Diesel Generator Cylinder Blocks" (FaAA Report No. FaAA-84-9-11.1) and dated December 1984. (Note that the FaAA report specifies additional inspections to be performed for blocks with "known" or "assumed" ligament cracks.) The inspection intervals (i.e., frequency) shall not exceed the intervals calculated using the cumulative damage index model in the subject FaAA report. In addition, inspection methods shall be consistent with or equivalent to those identified in the subject FaAA report.

Blocks determined in the future to have "ligament" cracks as the result of the above inspections should be inspected at each refueling outage to determine whether or not cracks have initiated on the top surface, which was exposed because of the removal of two or more cylinder heads. This process should be repeated over several refueling outages until the entire block has been inspected. If after this process has been completed new "ligament" cracks are found, this process should again be repeated. Liquid penetrant testing or a similarly sensitive non-destructive testing technique should be used as appropriate to determine the depth of any cracks discovered.

Whenever diesel generator No. 1 is operated in excess of 4,375 kW for one hour or more, a visual inspection of the right bank cylinder block is to be performed under intense light within 48 hours after engine shutdown to verify the absence of "stud-to-stud" and "stud-to-end" cracks.

If "stud-to-stud" or "stud-to-end" cracks are found, these findings are to be reported to the NRC within 24 hours. The affected engine is to be considered inoperable and is not to be restored to OPERABLE status until the disposition and/or corrective actions have been approved by the NRC staff.

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[3]This report was transmitted to H.R. Denton, (NRC), from C.L. Ray, Jr., (TDI Owners Group), by letter dated December 11, 1984.

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.

4.6 RADIOACTIVE GASEOUS EFFLUENTS

4.6.1 DOSE RATE

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose rate due to the discharge of radioactive gaseous effluents is maintained within 10 CFR 20 limits.

SPECIFICATION:

A. The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.16.1 in accordance with the methods and procedures of the ODCM.

B. The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.16.1 in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.6.1.1.

BASIS: This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive materials discharged in gaseous effluents will not result in the exposure of an individual in an UNRESTRICTED AREA, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 [10 CFR 20.106(b)]. For individuals who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above "background to an individual at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to ≤ 1500 mrem/year for the nearest cow to the plant.

4.6.3 DOSE, IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose due to iodine-131, iodine-133, tritium an radionuclies in particulate form with half-lives greater than 8 days is maintained as low as is reasonably achievable.

SPECIFICATION: Cumulative does contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the ODCM at least once per 31 days.

BASIS: This specificat´ n implements the requirements in Section III.A of Appendix I, 10 CFR Part 50, that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated.

The ODCM equations provided for determining the actual doses are based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation and subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

4.8 REACTIVITY ANOMALIES

APPLICABILITY: Applies to potential reactivity anomalies.

OBJECTIVE: To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION: A. Following a normalization of the computer boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made within 30 days and reported to the NRC pursuant to Specification 6.9.2.

BASIS: To eliminate expected errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a reactivity insertion of this amount would not result in pressure or temperature transients which exceed the design conditions of the reactor coolant system.

## 4.10 AUGMENTED INSERVICE INSPECTION OF HIGH ENERGY LINES OUTSIDE CONTAINMENT

APPLICABILITY: Applies to welds in piping systems or portions of systems located outside containment where protection from the consequences of postulated pipe breaks is not provided by a system of pipe whip restraints, protective enclosures, or other measures specifically designed to cope with such breaks.

OBJECTIVE: To provide assurance of the continued integrity of the piping systems over their service lifetime.

SPECIFICATION: A. For the welds in the main steam, main feedwater, and first point extraction lines identified in Reference 1, Table 1 and Table IA, Column: "Break Point Location", for which inservice inspection is specified in Column: "Solution",

(1) At refueling outage No. 4, a baseline inspection consisting of a volumetric examination of all specified welds shall be performed in accordance with the requirements of ASME Section XI Code, "Inservice Inspection of Nuclear Reactor Coolant Systems" 1971, up to and including 1972 addenda.

(2) Subsequent to the baseline examination, the inservice inspection of each weld shall be performed in accordance with the requirements of the Edition and Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" as required by Technical Specification 4.7, with the following schedule:

(a) During the first 3-1/3 years (or nearest refueling outage), volumetric examination of 100% all welds.

(b) Every 10 years thereafter (or nearest refueling outage), volumetric examination of 33-1/3% of the welds at the expiration of each 1/3 of the inspection interval with a cumulative 100% coverage of all welds every 10 years.

4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

APPLICABILITY: Applies to the leakage of radioactive source materials.

OBJECTIVE: To verify the physical integrity of portable and fixed radioactive calibration sources.

SPECIFICATION:

A. Byproduct material sealed sources which exceed the quantities listed in 10 CFR 30.71, Schedule B, and all other sealed sources containing greater than 0.1 microcuries shall be leak tested in accordance with Specification B, C and D below.

Exception: Not withstanding the periodic leak test required by this specification, any licensed sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

B. Each sealed source containing radioactive material, other than Hydrogen 3, with a half life greater than thirty days and in any form other than gas, shall be tested for leakage and/or contamination prior to use out of storage and prior to transfer to another person and thereafter at intervals not to exceed six months. This test does not apply to sealed sources that are stored and not in use.

C. The leakage test shall be capable of detecting the presence of .005 microcuries of radioactive material. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate.

D. If testing reveals the presence of .005 microcuries or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or disposed of in accordance with applicable regulatory requirements and reported in the subsequent annual report filed pursuant to Specification 6.9.1.4.

BASIS: This Specification assures that leakage from radioactive material sources does not exceed allowable total body or organ limits. In the unlikely event that those quantities of radioactive byproduct materials of interest to this Specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39 (c) limits for plutonium.

4.16 INSERVICE INSPECTION OF STEAM GENERATOR TUBING

APPLICABILITY: Applies to the inservice inspection and sampling selection for steam generator tubing.

OBJECTIVE: To monitor the integrity of the steam generator tube primary boundary and provide guidance for corrective action when imperfections are observed.

SPECIFICATION:

A. GENERAL STEAM GENERATOR TUBE SELECTION

The steam generators shall be inspected when shutdown by selecting steam generator tubes on the following basis:

1. Tubes for the inspection shall be selected on a random basis except where experience at San Onofre Unit 1 or experience in similar plants indicates critical areas to be inspected.

2. Each inspection shall include at least 3 percent of the total number of tubes in each steam generator to be inspected.

3. Inservice inspections may be limited to one steam generator on a rotating schedule encompassing 3 percent of the total tubes of steam generators in the plant if the results of previous inspections indicate that all steam generators are performing in a like manner.

4. Every inspection shall include all non-plugged tubes in the steam generator(s) to be inspected that previously had detected imperfections greater than 20 percent.

B. SUPPLEMENTARY INSPECTIONS

If the inspections in Specification A indicate imperfections, additional inspections shall be required as follows:

1. If any of the tubes inspected pursuant to Specification A.3 are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration since their last inspection, inspect 3 percent of the tubes in one of the uninspected steam generators.

2. If more than 10 percent of the tubes inspected in a steam generator are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration since their last inspection, or one or more of the tubes inspected have an imperfection in excess of the plugging limit, inspect an additional 3 percent of the tubes in that steam generator, concentrating on tubes in those areas of the tube sheet array where tubes with imperfections were found and on that length of tube where the imperfections were found. In addition, the rest of the steam generators shall be inspected in accordance with Specification A.2.

3. If the additional inspection in Specification B.2 indicates that more than 10 percent of the additionally inspected tubes are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration since their last inspection, or one or more of the additionally inspected tubes have an imperfection in excess of the plugging limit, inspect an additional 6 percent of the tubes in that steam generator in the area of the tubesheet array where tubes with imperfections were found and through that length of tube where the imperfections were found.

C. INSPECTION FREQUENCY

The inspections in Specifications A and B above shall be performed at the following frequencies:

1. Inservice inspections shall be not less than 10 nor more than 24 calendar months after the previous inspection.

2. If two consecutive inspections indicate that less than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, the inspections shall be not less than 10 nor more than 40 calendar months after the previous inspection.

3. Unscheduled inspections shall be conducted in accordance with Specification A in the event of primary-to-secondary leaks exceeding Specification 3.1.4 a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

D. ACCEPTANCE CRITERIA

1. As used in this specification:

   a. Imperfection means an exception to the dimensions, finish, or contour required by drawing or specification. Detectable eddy current testing indications below 20 percent of the nominal tube wall thickness, may be considered as imperfections.

   b. Degradation means a service-induced cracking, wastage, pitting, wear or general corrosion occurring on either inside or outside of a tube.

   c. Degraded Tube means a tube containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation above the tube roll expansion. Also, a tube with an imperfection in the region one-half inch below the uppermost one inch of sound roll, is considered a degraded tube.

   d. Defect means an imperfection of such severity that it exceeds the plugging limit or an imperfection located in the uppermost one inch of the tube roll expansion. A tube or sleeve containing a defect is defective.

e. Plugging Limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50 percent of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 40 percent of the nominal sleeve wall thickness.

For the tube roll expansion region, the following criteria apply:

(i) Any tube containing an imperfection within the uppermost one inch of sound roll shall be considered defective.

(ii) Any imperfection is acceptable below the uppermost one inch of sound roll.

f. Tube Roll Expansion means that portion of the tube which has been increased in diameter by a rolling process.

g. Sound Roll means a tube roll expansion which is fully expanded such that no crevice exists between the outside diameter of the tube and the tubesheet.

2. If, in the inspections performed under Specification A,

a. Less than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, and

b. No tube inspected exceeds the plugging limit,

plant operation may resume.

3. If, in the inspections performed under Specification B,

a. Less than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, and

b. No more than 3 of the tubes inspected exceed the plugging limit,

plant operation may resume after performance of the corrective action in Specification F.

4. If, in the inspections performed under Specification B,

a. More than 10 percent of the tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, or

b. More than 3 of the tubes inspected exceed the plugging limit, prior to resumption of plant operation, the situation shall be reported in a Special Report to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

5. The results of inspections performed under specifications A or B for all tubes in service which have defects below the uppermost one inch of tube roll expansion shall be reported to the Commission in a Special Report pursuant to Technical Specification 6.9.2 at least seven days prior to the resumption of plant operation. The report shall include:

   a. Identification of applicable tubes, and

   b. Location and size of the degradation.

E. CORRECTIVE ACTION

All leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limit shall be repaired or plugged.

BASIS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.3 gallons per minute per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.3 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require shutdown during which the leaking tubes will be located and plugged and additional inspections performed.

If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 50 percent of the tube nominal wall thickness, except where sleeves are installed, in which case the plugging limit is 40 percent of the nominal sleeve wall thickness. A plugging limit of 50 percent for tubes and 40 percent for sleeves ensures that defects will not occur between inspection intervals. Tubes with defects below the uppermost one inch of a sound roll may remain in service, provided there are no imperfections in this portion of the tube. The distance of one inch includes a 0.25 inch eddy current measurement uncertainty.

The results of tube ID gauging and dent detection conducted in San Onofre Unit 1 steam generators demonstrate that the denting process has been arrested. Continued assurance of this condition will be provided by monitoring for dent progression as part of the general steam generator tube inspection in accordance with Specification A. Progression of denting is adequately monitored in either steam generator A or C by reviewing required eddy current probe size reductions during the performance of this inspection scope.

The results of AVB area inspections conducted in San Orofre Unit 1 steam generators demonstrate that AVB modifications installed during the Cycle VI refueling outage were successful in eliminating significant growth of tube wall penetration indications at AVB locations. Continuing assurance of this condition can be provided by performing U-bend inspections as part of the Specification A inspection scope at refueling outage intervals of tubes having wall penetration indications in excess of 20 percent at AVB locations.

TABLE 4.18.1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

| Analysis | Water (pCi/l) | Airborne Particulate or Gas (pCi/m$^3$) | Marine Animals (pCi/kg, wet) | Local Crops (pCi/kg, wet) | Sediment (pCi/kG, dry) |
|---|---|---|---|---|---|
| gross beta | 4 | $1 \times 10^{-2}$ | | | |
| H-3 | 2000 | | | | |
| Mn-54 | 15 | | 130 | | |
| Fe-59 | 30 | | 260 | | |
| Co-58, 60 | 15 | | 130 | | |
| Zn-65 | 30 | | 260 | | |
| Zr-95 | 30 | | | | |
| Nb-95 | 15 | | | | |
| I-131 | 1[b] | $7 \times 10^{-2}$ | | 60 | |
| Cs-134 | 15 | $5 \times 10^{-2}$ | 130 | 60 | 150 |
| Cs-137 | 18 | $6 \times 10^{-2}$ | 150 | 80 | 180 |
| Ba-140 | 60 | | | | |
| La-140 | 15 | | | | |

c. A health physics technician# shall be on-site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent duties during this operation.

e. A Fire Brigade of at least five members shall be maintained on site at all times.# The Fire Brigade shall not include the Shift Superintendent and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff in the following job classifications:

   1) Shift Superintendents, Control Room Supervisors, Control Operators, Assistant Control Operators, Nuclear Plant Equipment Operators, Plant Equipment Operators;

   2) Electricians and their first line supervisors;

   3) I&C Technicians, Computer Technicians, Test Technicians and their first line supervisors;

   4) Operational Health Physics Technicians and their first line supervisors;

   5) Boiler and Condenser Mechanics, Machinists, Welders, Crane Operators and their first line supervisors;

   6) Contractor or other Department personnel performing functions identical to those performed by personnel identified in items 1 through 5 above and within the organizational framework of the station.(1)

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel identified above work a normal 8-hour day, 40-hour week (excluding shift turnover and meal time) while the plant is operating (MODES 1, 2, 3 and 4). However, in the event that

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\# The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

(1) Shift Technical Advisors are exempt from the overtime guidelines specified, since sleeping accommodations are provided.

## 6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager, Nuclear Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager, Station Emergency Preparedness and shall meet or exceed the requirements of National Fire Protection Association Standard No. 27-1975.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION

APPENDIX A

TECHNICAL SPECIFICATIONS

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# FIGURES

# SAN ONOFRE NUCLEAR GENERATING STATION, UNIT 1
## TECHNICAL SPECIFICATIONS
## TABLES

TABLES

SHUTDOWN MARGIN

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

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

SOLIDIFICATION

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

SOURCE CHECK

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

STAGGERED TEST BASIS

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

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

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

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 adsorbers 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

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.

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 order to meet acceptance criteria for certain secondary side transients, the pressurizer high level trip must be set at 50% level or less.(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, 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 condition 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)

3. Steam/Feedwater Flow Mismatch

A significant mismatch of steam flow and feedwater flow to the steam generators occurs at greater than 50% power in the event of LONF and FLB. In the event of these transients, the 2 out of 3 mismatch trip logic will result in reactor trip on the order of 1 second after the initiating event. The safety analysis conservatively assumed that reactor trips would occur at 5 seconds and 10 seconds for LONF and FLB, respectively. The high and low settings assure that regardless of the type of mismatch occurring for individual loops, a protective reactor trip is provided, which satisfies the single failure criterion for the postulated events.(8)

4. Reactor Coolant Pump Breaker Open

The Reactor Coolant Pump (RCP) Breaker Open reactor trip provides a redundant trip to the low flow trip. The overcurrent trip of the RCP breakers protects the core following a locked rotor and the undercurrent trip of the RCP breakers protects the core following a sheared shaft. The trip settings are selected to meet the analysis assumption that rods begin to drop 6.1 seconds after the initiating event. The Reactor Protection System Permissives change the trip on RCP breaker open to 2/3 loops instead of 1/3 loops at power levels below 50%.

TABLE 2.1

MAXIMUM SAFETY SYSTEM SETTINGS

| | Three Reactor Coolant Pumps Operating |
|---|---|
| 1. Pressurizer High Level | ≤ 50% Level |
| 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 (0.394 ΔT+T avg.) - 14341 |
| **5. Coolant Flow | ≥ 85% of indicated full loop flow |
| ***6. Steam/Feedwater Flow Mismatch | |
| a. Low+ Setting: | $\frac{\text{Steam Flow - Feedwater Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$ |
| b. High+ Setting: | $\frac{\text{Feedwater Flow - Steam Flow}}{\text{Feedwater Flow @ 100\% Power}} \leq 0.25$ |
| **7. Reactor Coolant Pump Breaker Open | |
| a. Overcurrent | ≤ 2400 amps |
| b. Undercurrent | ≥ 110 amps |
| c. Undervoltage | ≥ 60% of rated bus voltage |

* The nuclear overpower trip 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.

***May be bypassed at power levels below 50% of full power.

\+ High and Low feedwater flow relative to steam flow.

BASIS:

Specification 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and

## 3.1.3 COMBINED HEATUP, COOLDOWN AND PRESSURE LIMITATIONS

APPLICABILITY: Applies to heatup and cooldown of the reactor coolant system.

OBJECTIVE: To maintain the structural integrity of the reactor coolant system throughout the lifetime of the plant.

SPECIFICATION:

A. Reactor pressure and heatup and cooldown of the reactor coolant system during the first 16 years of equivalent full power operation shall be limited in accordance with Figures 3.1.3a and 3.1.3b. Thereafter, limits shall be based on neutron exposure equivalent to not less than 16 years of full power operation, and Figures 3.1.3a and 3.1.3b shall be updated accordingly (by formal license amendment application).*

B. Figures 3.1.3a and 3.1.3b shall be updated in accordance with the following criteria and procedures:

(1) The methods of Appendix G, "Protection Against Nonductile Failure", to Section III of the ASME Boiler and Pressure Vessel Code shall be used to obtain the allowable pressure-temperature relationships for the reactor coolant system.

(2) The curves in Figure 3.1.3c shall be used in predicting the Reference Nil-Ductility Temperature ($RT_{NDT}$) increase, unless measurements on the irradiation specimens show $RT_{NDT}$s greater than those predicted by the curves, in which case a new curve having the same slope as the original shall be constructed.

C. The pressurizer heatup rate of 100°F/hour and cooldown rate of 200°F/hour shall not be exceeded.

D. The reactor shall not be brought to a critical condition until the pressure-temperature state is to the right of the criticality limit line as shown in Figure 3.1.3a.

BASIS: The initial Reference Nil-Ductility Temperature ($RT_{NDT}$) for all reactor vessel material based on Charpy V-notch data, drop weight tests, and conservative estimates** is 82°F or less. The $RT_{NDT}$ at the 1/4 thickness location (location of Appendix G reference flaw tip) increases as a function of cumulative neutron exposure up to approximately 240°F for the core region of the reactor vessel after 30 years of operation.

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* Technical Specification 3.20.A(1) should be reevaluated for continued applicability of the low pressure PORV overpressure setpoint at any time the heatup and cooldown curves are changed.

** NRC Standard Review Plan Branch Technical Position MTEB 5-2.

A sixteen (16) equivalent full power year service period was chosen for the operational limits given in this specification because at the end of this period the limiting $RT_{NDT}$ of the reactor vessel at the 1/4 thickness location is approximately 217°F in the core region. This $RT_{NDT}$ is at least 50°F above the $RT_{NDT}$ of all other regions in the primary reactor coolant system.

The highest $RT_{NDT}$ of the core region material is determined by adding the radiation induced $\Delta RT_{NDT}$ for the applicable time period to the original $RT_{NDT}$ shown in Table 3.1.3.1. The fast neutron (E > 1Mev) fluence at 1/4 thickness and 3/4 thickness vessel locations is given as a function of full power service life in Figure 3.1.3d. Using the applicable fluence at the end of the year period and the copper content of the material in question, the RTNDT is obtained from Figure 3.1.3c.

Values of $\Delta RT_{NDT}$ may continue to be determined in this manner unless measurements on the irradiation specimens show $\Delta RT_{NDT}$s greater than those predicted by the curves for the equivalent capsule exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from non-mandatory Appendix G in Section III of the ASME Boiler and Pressure Vessel Code, and discussed in detail in Reference 1.

The results of these calculations are provided in Reference 2.

The design heatup and cooldown rates for the pressurizer are 100°F/hour and 200°F/hour, respectively.

The vertical line portion of the criticality limit given in Figures 3.1.3a is at the minimum permissible temperature for the 2485 psig in-service hydrostatic test as required by Appendix G to 10CFR Part 50. The non-vertical portion of the criticality limit is shifted 40°F to the right of the heatup curve as required by Appendix G to 10CFR Part 50.

REFERENCES: (1) "Pressure Temperature Limits" Section 5.3.2 of Standard Review Plan, NUREG-751087, 1975.

(2) S. E. Yanichko, et al, "Analysis of Capsule F from the Southern California Edison Company San Onofre Reactor Vessel Radiation Surveillance Program", WCAP 9520, May 1979.

(2) Containment Spray System

a. Two refueling water pumps are OPERABLE.

b. Two hydrazine additive pumps are OPERABLE.

c. Hydrazine tank level and hydrazine concentration comply with Specification 3.3.4.

(3) Valves and interlocks associated with each of the above systems are OPERABLE.

(4) Effective leakage from the recirculation loop outside the containment shall be less than 625 cc/hr as calculated from the following formula.

Effective Leakage = $a_1 \times L_1 + a_2 \times L_2 + a_3 \times L_3$

where,

$L_1$ = pump and valve leakage which drains to auxiliary building sump

$L_2$ = valve leakage in auxiliary building or doghouse

$L_3$ = valve leakage outside

$a_1$ = iodine release factor for leakage in auxiliary building sump

$a_2$ = iodine release factor for leakage in auxiliary building or doghouse

$a_3$ = iodine release factor for leakage outside the auxiliary building or doghouse

If effective leakage from the recirculating loop outside the containment exceeds 625 cc/hr, make necessary repairs to limit leakage to 625 cc/hr within 72 hours or be in COLD SHUTDOWN within the next 36 hours.

B. During critical operation or when the reactor coolant system temperature is above 200°F, as appropriate per Item A above, maintenance shall be allowed on any one of the following items at any one time:

(1) One motor-operated valve at a time (MOV 1100B or 1100D) in the recirculation loop upstream of the charging pump suction header for a period of time not longer than 72 consecutive hours.

(2) One refueling water pump and/or its associated discharge valve at a time, for a period not longer than 72 consecutive hours.

(3) One hydrazine pump and/or its associated discharge valve (SV600 or 601) at a time, for a period of time not longer than 72 consecutive hours.

(4) One charging pump for a period of time not longer than 72 consecutive hours.

(5) One of the two required component cooling water pumps for a period of time not longer than 72 consecutive hours.

(6) One of the two saltwater cooling pumps with the auxiliary saltwater cooling pump or screen wash pumps available as backup for a period of time not longer than 72 consecutive hours. The backup pump(s) shall be demonstrated operable by test within 1 hour of declaring the saltwater pump inoperable.

(7) One train of ESF switchover automatic trip for a period of time not to exceed 72 consecutive hours.

(8) One motor-operated valve at a time (MOV-1100C or MOV-1100E) in the VCT outlet line to the charging pump suction for a period of time not longer than 72 consecutive hours.

C. Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to demonstrate availability.

D. In the event of a failure of a recirculating pump, plant operation may continue provided operability of the remaining pump and its associated motive and control power are satisfactorily demonstrated on a daily basis, including verification that the containment spray bypass valves (CV517 and 518) are closed.

<u>BASIS</u>: The requirements of Specification A assure that before the reactor can be made critical, or before the reactor coolant system heatup is initiated, adequate engineered safeguards are OPERABLE. The limit of 625 cc/hr for the recirculation loop leakage ensures that the combined 0-2 hr EAB thyroid dose due to recirculating loop leakage and containment leakage will not exceed the limits of 10 CFR 100. The formula for determining the leakage incorporates consideration of the significance of leakage in different plant areas. The iodine release factor adjusts actual pump or valve leakage to account for the fraction of the iodine in the leakage which would actually be released to the atmosphere. The iodine release factors in the auxiliary building sump, the auxiliary building or doghouse, and outside are 0.05, 0.5, and 1.0, respectively.

When the reactor is critical or the reactor coolant system temperature is above 200°F, maintenance is allowed per Specifications B and C providing requirements in Specification C are met which assure OPERABILITY of the redundant component. The specified maintenance times are a maximum, and maintenance work will proceed with diligence to return the equipment to an operable condition as promptly as possible. OPERABILITY of the specified components shall be based on the results of Specification No. 4.2.

The allowable maintenance periods are based upon the repair of certain specific items. Based on the demonstration that equipment redundant to that removed from service is OPERABLE, it is reasonable to maintain the reactor at power over this short period of time.

In the unlikely event that the need for safety injection should occur:

-- functioning of one train will protect the core.[1][2][3][4] Containment sprays alone, however, will maintain containment pressure under design pressure.[5]

-- functioning of one of the two hydrazine additive pumps and associated discharge valve will effect introduction of hydrazine into containment spray water. This provides for adsorption of airborne fission products and reduction of the thyroid doses associated with the maximum hypothetical accident to within 10 CFR 100 limits.

-- dissolution of 5400 pounds of anhydrous trisodium phosphate stored in the sump will ensure that the pH of the water in the sump will be greater than 7 within four (4) hours, so as to prevent chloride stress corrosion cracking of systems and components exposed to the circulating sump water.

In the event of inoperability of a recirculation pump, plant operation may continue since either pump is sufficient and a daily OPERABILITY demonstration of the remaining pump and its associated motive and control power provides assurance that it will be OPERABLE if required.

The switchover from injection to recirculation modes is a two part process, which consists of the automatic termination of the flow from SI/FW pumps including automatic pump trip and automatic closures of MOV's 850 A, B and C followed by manual realignment to recirculation from the containment sump. The automatic trip setpoint is bounded by the minimum water level in the sump to support recirculation for long term post-LOCA cooling and the minimum RWST level to support charging and containment spray during the manual realignment. The setpoint analysis conservatively determined the automatic trip setpoint to be 20% of the RWST level. The automatic trip setpoint is the result of the combination of the worst single active failure considering SIS and SISLOP conditions.

3.3.2 SHUTDOWN STATUS

APPLICABILITY: Applies to piping connections between the feedwater condensate system and the reactor coolant system.

OBJECTIVE: To preclude injection of feedwater condensate into the reactor coolant system when the reactor is shut down and to preclude the potential for overpressurization when water solid.

SPECIFICATION: A. When reactor fuel assemblies are in the vessel and the reactor coolant pressure is less than 500 psig, two "positive barriers" shall be provided between the feedwater condensate system and the piping connections to the reactor coolant system. Additionally, when the reactor coolant system is water solid at less than 500 psig, two positive barriers shall be provided between the safety injection system and piping connections to the reactor coolant system. A "positive barrier" is defined as follows:

(1) Motor Operated Valves

When closed and tagged with supply breakers open, except that power may be restored during no-flow tests of the safety injection system (Specification No. 4.2).

(2) Pneumatic/Hydraulic Operated Valves

When closed and the condition tagged with the respective hydraulic block valve closed except that they may be opened during no-flow tests of the safety injection system (Specification No. 4.2).

(3) Manually Operated Valves

When closed and condition tagged.

(4) Feedwater Pump (Overpressurization Protection Only)

When shutdown with the breaker in the racked out condition.

BASIS: Under normal conditions, system operational interlocks assure that injection of feedwater condensate to the reactor by

## 3.3.4 MINIMUM SOLUTION VOLUME HYDRAZINE CONCENTRATION IN THE HYDRAZINE TANK

APPLICABILITY: Applies to the inventory of spray additive solution.

OBJECTIVE: To insure availability of containment spray additive solution of required quality.

SPECIFICATION: When the reactor coolant system temperature is above 200·F, the hydrazine tank shall contain not less than 150 gallons of aqueous solution having a concentration of not less than 21 wt% $N_2H_4$.

BASIS: The hydrazine tank serves the purpose of acting as a reservoir of aqueous hydrazine solution for post-accident iodine removal.

100 gallons of $N_2H_4$ solution are required to reduce airborne iodine concentration in the event of a loss of coolant accident. By adding a 50% margin to this figure to ensure that NPSH to the spray addition pumps is maintained at all times, a total of 150 gallons is required. This amount fulfills requirements for safety injection operations.

3.4.2 MAXIMUM SECONDARY COOLANT ACTIVITY

APPLICABILITY: Applies to measured maximum radioiodine activity in the secondary coolant of the steam generators any time the primary coolant system temperature exceeds 200·F.

OBJECTIVE: To limit the consequences of an accidental release of secondary coolant to the environment.

SPECIFICATION: A. The specific activity of radioiodine in the secondary coolant shall be limited to 0.1 mCi/gm DOSE EQUIVALENT I-131.

ACTION: B. With the specific activity of the secondary coolant in excess of 0.1 mCi/gm DOSE EQUIVALENT I-131, the reactor shall be placed in cold shutdown within 36 hours.

BASIS: The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The restriction of 0.1 mCi/gram DOSE EQUIVALENT I-131 in the secondary system limits the 2 hour thyroid exposure dose to well within the guidelines of 10 CFR Part 100 at the site boundary under these accident conditions. This thyroid dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analysis.

The assumptions and results of these calculations are documented in "Safety Evaluation by the Office of Nuclear Reactor Regulation," Docket No. 50-206, dated April 1, 1977.

## 3.4.3 AUXILIARY FEEDWATER SYSTEM

APPLICABILITY: Applies to the auxiliary feedwater pumps and valves for MODES 1, 2 and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat from the core.

SPECIFICATION: Two trains of auxiliary feedwater including associated pumps and valves, shall be OPERABLE.

ACTION:

A. With one Train of auxiliary feedwater inoperable, restore the inoperable train to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

B. With both Trains of auxiliary feedwater inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of offsite power.

Two auxiliary feedwater trains and the steam system relief valves provide core decay heat removal capability in the event of a sustained loss of offsite power. Either auxiliary feedwater train has the capability to satisfy decay heat removal requirements from the core, with a delivered flow of at least 185 gpm per train with three intact main feedwater lines and pressurized steam generators, 100 gpm per train with two intact main feedwater lines and pressurized steam generators, and 175 gpm per train with two intact main feedwater lines and depressurized steam generators.

AFW System Train A pumps and valves consist of AFW pumps G-10S and G-10 and associate valves, including flow control valves FCV-2300A, FCV-2300B, and FCV-2300C.

AFW System Train B pump and valves consist of AFW pump G-10W and associated valves, including flow control valves FCV-3300A, FCF-3300B, and FCV-3300C.

3.4.4 AUXILIARY FEEDWATER STORAGE TANK

APPLICABILITY: Applies to the auxiliary feedwater storage tank for MODES 1, 2 and 3.

OBJECTIVE: To ensure the availability of auxiliary feedwater to remove decay heat.

SPECIFICATION: A. The auxiliary feedwater storage tank (AFST) shall be OPERABLE with a usable water volume of at least 190,000 gallons of water.

ACTION: B. With the AFST inoperable, within 4 hours restore the AFST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

BASIS: The OPERABILITY of the auxiliary feedwater storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions (including cooldown) for 32 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. In addition, the water volume will provide sufficient margin to account for spillage that occurs during a main feedwater line break with loss of AFW flow indication prior to isolation of the broken line. Spillage is assumed to last no longer than one hour until the broken loop is identified via RCS Loop Delta-T positive indication that will be evident for the two intact steam generators. The usable water volume limit is specified relative to the bottom of the tank indicated level range (i.e., level tap). The contained water volume below this datum provides a significant margin to the NPSH and vortexing limits above the highest AFW pump suction inlet in the tank, but is not considered available for purposes of this specification.

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 with the SETPOINTS and RESPONSE TIMES as shown in Tables 3.5.1-2 and 3.5.1-3, respectively.

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 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

| FUNCTION UNIT | | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 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##### | 6# |
| 12. | Turbine Trip-Low Fluid Oil Pressure | 3 | 2 | 2 | 1#### | 6# |
| 13. | Reactor Coolant Pump Breaker Position | | | | | |
| | 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# |
| 14. | 4kV Bus 1C and Bus 2C Undervoltage | 2/bus | 1/bus from both buses | 1/bus from both buses | 1,2,3*,4* | 1# |

2) The SHUTDOWN MARGIN BASIS of Specification 3.5.2 is determined at least once per 12 hours.

3) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours.

4) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or

5 ) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$35 steps of the inoperable rod within one hour while maintaining the rod insertion limits of Figure 3.5.2.1.

<u>BASIS</u>: The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses.

The misalignment allowance of Specification B, assures core performance within allowed design margins including allowance for the inaccuracy of the position signals.

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* | | |
| o Auxiliary Feedwater Flow Rate | 1/steam generator | 1/steam generator |
| o Steam Generator Water Level (Wide Range) | 1/steam generator | 1/steam generator |
| o Reactor Coolant System Loop Delta-T Indication | 1/loop | 1/loop |
| 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 |
| Refueling Water Storage Tank Level | 2 | 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 is provided by one channel of auxiliary feedwater flow rate (Train B), one channel of environmentally qualified steam generator wide range level (Train A), and one channel of RCS Loop Delta-T indication. These comprise the three types of indication of auxiliary feedwater flow for each steam generator.

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

## TABLE 3.5.7-1

## AUXILIARY FEEDWATER INSTRUMENTATION

| FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|
| a. Manual Actuation | 2 | 1 | 2 | 1, 2, 3 | 12 |
| b. Automatic Actuation Logic | 2 | 1 | 2 | 1, 2, 3 | 13 |
| c. Steam Generator Water Level-Low | | | | | |
| i. Train A | 3 | 2 | 2 | 1, 2, 3 | 14, 15 |
| ii. Train B | 3 | 2 | 2 | 1, 2, 3 | 14, 15 |
| d. AFW Train Interlocks* | | | | | |
| i. Low Flow Train B/ Start Train A Flow | | | | | |
| 1) Start Pump G10S/Open Pump G10 Discharge Valve CV-2620, AND | 2 | 1 | 2 | 1, 2, 3 | 35, 36 |
| 2) Start Pump G10/Open Pump G10S Discharge Valve MOV-1202 | 2 | 1 | 2 | 1, 2, 3 | 35, 36 |
| ii. Normal Flow Train B/ Stop Train A Flow | | | | | |
| 1) Stop Pump G10S/Close Pump G10 Discharge Valve CV-2620, OR | 2 | 2** | 2 | 1, 2, 3 | 35, 36 |
| 2) Stop Pump G10/Close Pump G10S Discharge Valve MOV-1202 | 2 | 2** | 2 | 1, 2, 3 | 35, 36 |

* A total of 4 flow switches monitor Train B flow and each switch represents a channel which provides the specified signals to Train A.

** Only 1 of 2 Channels is required to trip if 1 Channel has been disconnected per the requirements of ACTION 35.

## TABLE 3.5.8.1

## RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | ACTION |
|---|---|---|
| 1. Gross Radioactive Monitors Providing Automatic Termination of Release | | |
| a. Liquid Radwaste Effluent Line (R-1218) | (1) | 16 |
| b. Steam Generator Blowdown [a] Effluent Line (R-1216) | (1) | 17 |
| c. Turbine Building Sumps Effluent Line (Reheater Pit Sump) (R-2100) | (1) | 18 |
| d. Yard Sump (R-2101) | (1) | 18 |
| e. Component Cooling Water System [b] (R-1217) | (1) | 19 |
| 2. Flow Rate Measurement Devices | | |
| a. Liquid Radwaste Effluent Line (FE-16, FE-18) | (1) | 20 |
| b. Circulating Water Outfall* | | |
| c. Steam Generator Blowdown Effluent* Line | | |

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* Pump status, valve turns or calculations are utilized to estimate flow.

(a) Secondary coolant samples and activity analysis performed in accordance with T.S. 4.1, Table 4.1.2.

(b) Closed loop system. Monitor closes vent valve to isolate surge tank.

TABLE 3.5.8.1
(Continued)

TABLE NOTATION

ACTION 16 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

1. At least two separate samples which can be taken by a single person are analyzed in accordance with Specification 4.5.1.A., and;

2. At least two technically qualified persons verify the release rate calculations and discharge valving.

ACTION 17 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least $10^{-7}$ microcurie/ml;

1. At least once per 12 hours when the specific activity of the secondary coolant is > 0.01 µCi/gram DOSE EQUIVALENT I-131.

2. At least once per 24 hours when the specific activity of the secondary coolant is ≤ 0.01 µCi/gram DOSE EQUIVALENT I-131.

ACTION 18 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least $10^{-7}$ microcurie/ml.

ACTION 19 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, determine if there is leakage from the Component Cooling Water System to the Salt Water Cooling System. If leakage exists, sample the Component Cooling Water System to estimate the activity being released via the Salt Water Cooling System at least once per 24 hours for gross activity (beta or gamma) at a lower limit of detection of at least $10^{-7}$ microcurie/ml.

ACTION 20 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in-situ may be used to estimate flow.

## TABLE 3.5.10-1

## RADIATION MONITORING INSTRUMENTATION

| INSTRUMENT | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ALARM SETPOINT | MEASUREMENT RANGE | ACTION |
|---|---|---|---|---|---|
| 1. AREA MONITORS | | | | | |
| a. Control Room Area (R-1231) | 1 | All | 1 mR/hr | $10^{2}$ - $10^{2}$mR/hr | 25 |
| b. Spent Fuel Pool Area (R-1236) | 1 | * | 25 mR/hr | $10^{2}$ [illegible] | 26 |
| c. Containment Radiation Monitor-High Range (R-1255, R-1257) | 2 | 1, 2, 3 & 4 | 10 R/hr | 1-[illegible] | 27 |
| 2. PROCESS MONITORS | | | | | |
| a. Wide Range Gas Monitor (R-1254) | 1 | 1, 2, 3 & 4 | per ODCM | $10^{-}$-$10^{5}$ μCi/cc | 27 |
| b. Main Steam Dump and Safety Valve Channels (R-1256A&B, R-1258A&B) | 1/steamline | 1, 2, 3 & 4 | 1mR/hr (low)<br>1 R/hr (high) | $10^{1}$-$10^{4}$ mR/hr<br>$10^{1}$-$10^{4}$ R hr | 27 |

* With fuel in the spent fuel pool or building

| Δk/k | Event | Basis for Adequacy |
|---|---|---|
| 5% | Open reactor coolant system | Provides adequate margin so that maintenance activities can be carried out with the reactor head removed.[1] |

Regarding internal pressure limitations, the containment design pressure of 46.4 psig would not be exceeded if the sphere internal pressure before a major loss of coolant accident was no greater than 0.4 psig. The design criteria also allows an internal vacuum not in excess of 2.0 psig. Thus, the specified limiting conditions for internal pressure are consistent with the design basis.[2] Although such design values could be exceeded without damage to the structure, it is considered that the importance of the containment function warrants the specified values.

Opening of the ventilation system backup valves, CVS-301 and CVS-313, is not considered a violation of containment integrity during startup conditions provided that their corresponding in-line valves POV 9 and POV 10 are closed.

REFERENCES:

(1) Supplement No. 3 to Final Engineering Report and Safety Analysis, Question No. 2.

(2) Final Engineering Report and Safety Analysis, Paragraph 5.3.

3.6.2 CONTAINMENT ISOLATION VALVES

APPLICABILITY: MODES 1, 2, 3 and 4.

OBJECTIVE: To provide assurance that the containment isolation valves listed in Table 3.6.2-1 will function when initiated by appropriate sensors.

SPECIFICAIION: The containment isolation valves specified in Table 3.6.2-1 shall be OPERABLE.

ACTION:

A. With one or more of the isolation valve(s) specified in Table 3.6.2-1 inoperable, for each affected penetration that is provided with two isolation valves and is open, maintain at least one valve OPERABLE, and for all affected penetrations with either one or two isolation valves, one of the following Actions shall be taken:

1. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

2. Isolate each affected penetration within 4 hours by use of at least one deactivated* power operated valve secured in the isolation position, or

3. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

4. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. The provisions of Specification 3.0.4 are not applicable provided that within 4 hours the affected penetration is isolated in accordance with Action A.2 or A.3 above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are taken.

BASIS: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

* Valve may be temporarily activated for valve position verification and testing. While the valve is activated by this note, Action A.1 shall be applied and any system(s) declared inoperable pursuant to Action B shall not be declared OPERABLE.

## 3.7 AUXILIARY ELECTRICAL SUPPLY

### 3.7.1 ELECTRICAL SUPPLY: OPERATING

APPLICABILITY: MODES 1, 2, 3, and 4

OBJECTIVE: To define those conditions of electrical power availability necessary to provide for safe reactor operation and to provide for the continuing availability of engineered safeguards.

SPECIFICATION:

a. One Southern California Edison Company and one San Diego Gas & Electric Company high voltage transmission line to the switchyard and two transmission circuits from the switchyard, one immediate and one delayed access, to the onsite safety-related distribution system shall be OPERABLE. This configuration constitutes the two required offsite circuits.

b. Two redundant and independent diesel generators shall be OPERABLE each with total connected design load not to exceed 6,000 kW and with:

   1. A separate day tank containing a minimum of 290 gallons of fuel,

   2. A separate fuel storage system containing a minimum of 37,500 gallons of fuel, and

   3. A separate fuel transfer pump.

c. Train A Emergency AC Buses shall be OPERABLE, comprised of:

   1. 4,160 volt Bus 1C,
   2. 480 volt Buses 1 and 3, and associated station service transformers with tie breaker open.

d. Train B Emergency AC Buses shall be OPERABLE, comprised of:

   1. 4,160 volt Bus 2C,
   2. 480 volt Buses 2 and 4, and associated station service transformers with tie breaker open.

e. 120 volt AC Vital Buses 1, 2, 3, 3A, and 4 energized from associated inverters connected to DC Bus 1.

f. 120 volt AC Vital Buses 5 and 6 energized from associated inverters connected to DC Bus 2.

g. 125 volt DC Bus 1 shall be OPERABLE and energized from Battery No. 1, with at least one full capacity charger.

h. 125 volt DC Bus 2 shall be OPERABLE and energized from Battery No. 2, with at least one full capacity charger.

i. Two trains of Safeguards Load Sequencing Systems (SLSS) shall be OPERABLE.*

j. The MOV-850C Uninterruptible Power Supply (UPS) shall be OPERABLE and energized from the battery with its full capacity charger.**

k. Manual Transfer Switch 7 (MTS-7) shall be OPERABLE and energized from MCC-2.

l. Manual Transfer Switch 8 (MTS-8) shall be OPERABLE and energized from MCC-4.

ACTION:

A. With one of the required offsite circuits inoperable, demonstrate the operability of the remaining AC sources by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 8 hours thereafter and Surveillance Requirement B.1.a within 24 hours. Restore the circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

B. If one diesel generator is declared inoperable, demonstrate the operability of the two offsite transmission circuits and the remaining diesel generator by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 8 hours thereafter and Surveillance Requirement B.1.a within 24 hours. Restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

C. With one offsite circuit and one diesel generator of the above required AC electrical power sources inoperable, demonstrate the operability of the remaining AC sources by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 8 hours thereafter and Surveillance Requirement B.1.a within 8 hours. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Have at least two offsite circuits and two diesel generators OPERABLE within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

D. With one diesel generator inoperable as in B or C above, verify that: (1) all required systems, subsystems, trains,

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* The automatic load function may be blocked in Mode 3 at a pressurizer pressure ≤ 1900 psig.

** Applicable in MODES 1, 2, and 3 above 500 psig.

components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE; and (2) the steam-driven auxiliary feedwater pump is OPERABLE in MODES 1, 2, and 3. If these conditions are not satisfied within 2 hours, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

E. With two required offsite circuits inoperable, demonstrate the operability of two diesel generators by performing Surveillance Requirement B.1.a of Technical Specification 4.4 within 8 hours, unless the diesel generators are already operating. Restore at least one of the inoperable sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 4 hours. Have at least two offsite circuits and two diesel generators OPERABLE within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

F. With two of the above required diesel generators inoperable, demonstrate the operability of two offsite circuits by performing Surveillance Requirement A of Technical Specification 4.4 within one hour and at least once per 2 hours thereafter. Restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

G. With less than the above trains of Emergency AC buses OPERABLE, restore the inoperable buses within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

H. With one AC Vital Bus either not energized from its associated inverter, or with the inverter not connected to its associated DC Bus: (1) re-energize the AC Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) re-energize the AC Vital Bus from its associated inverter connected to its associated DC Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

I. With one DC bus inoperable or not energized from its associated battery and at least one full capacity charger, re-energize the DC Bus from its associated battery and at least one full capacity charger within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

J. With one Safeguards Load Sequencing System inoperable, restore the inoperable sequencer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

K. With the MOV-850C UPS inoperable, or not energized from its associated battery and its full capacity charger, restore the UPS to OPERABLE status and re-energize the UPS from its associated battery and its full capacity charger within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

L. With MTS-7 inoperable or not energized from MCC-2, restore MTS-7 to OPERABLE status and re-energize MTS-7 from MCC-2 within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

M. With MTS-8 inoperable or not energized from MCC-4, restore MTS-8 to OPERABLE status and re-energize MTS-8 from MCC-4 within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

BASIS: The station is connected electrically to the Southern California Edison Company and San Diego Gas & Electric Company system via either of two physically independent high voltage transmission routes composed of four Southern California Edison Company high voltage lines and four San Diego Gas & Electric Company high voltage lines.

Of the four Southern California Edison Company lines, any one can serve as a source of power to the station auxiliaries at any time. Similarly, any of the four San Diego Gas & Electric Company lines can serve as a source of power to the station auxiliaries at any time. By specifying one transmission line from each of the two physically independent high voltage transmission routes, redundancy of sources of auxiliary power for an orderly shutdown is provided.

Similarly, either transformer A or B, along with transformer C, provide redundancy of 4160 volt power to the auxiliary equipment, and in particular to the safety injection trains. In addition, each 4160 volt bus has an onsite diesel generator as backup.

In MODES 1, 2, 3 and 4, two diesel generators provide the necessary redundancy to protect against a failure of one of the diesel generator systems or in case one diesel generator system is taken out for maintenance, without requiring a reactor shutdown. This also eliminates the necessity for depending on one diesel generator to operate for extended periods without shutdown if it were required for post-accident conditions.

When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE. In addition, the ACTION STATEMENT requires a verification that the steam-driven auxiliary feedwater pump is OPERABLE in MODES 1, 2, and 3.

These requirements are intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the operability of the component. The Safeguards Load Sequencing System is designed so that each sequencer starts and loads its associated diesel generator and sequences the ECCS loads upon receipt of a safety injection signal (SIS) and concurrent loss of voltage on its respective 4160 volt bus (i.e., upon a SISLOP).

The total connected design load on each diesel generator is restricted to 6,000 kW or less. This requirement was the result of a crankshaft crack propagation analysis (See Reference 1). The analysis postulated that the crankshaft initially has stress-induced surface cracks. The analysis then considered the effect of four types of diesel load histories on the growth of these cracks. Each load history consisted of repeated start-stop cycles with some steady state operation at full load (6,000 kW) between each start and its stop. The analysis concluded that for a crankshaft with a detectable size crack (10 mils deep), the number of start-stop cycles required to enlarge the crack until it becomes self-propagating (18 mils deep) under the full load steady state stresses represents the effective life of the crankshaft.

During normal operations, the 480 volt system is considered OPERABLE if the four 480 volt buses and four station service transformers are OPERABLE with respective tie breakers open. This will ensure that the 480V main breakers and transformers remain OPERABLE during the worst loading condition in case of a SIS without LOP.

The primary power source for Vital Buses 1, 2, 3, 3A, and 4 is Train A DC Bus 1. The alternate power source is available from MCC-2 through MTS-7. The 1987 RPS and ESF single failure analyses credited the Train B backup power to these vital buses through MTS-7.

Correct operation of the safety injection system is assured by the operability of the load sequencers and the UPS for MOV-850C and MOV-358 (MOV-850C UPS). Correct operation of the recirculation system is assured by the operability of the MOV-850C UPS which also supplies MOV-358.

Manual Transfer Switch 8 (MTS-8) provides the means to power MOV-883 and the MOV-850C UPS from either Train A or Train B. However, due to single failure considerations and environmental effects, MTS-8 is normally powered from MCC-4 on Train B. MOV-883 is the discharge valve from the RWST and must remain open during the safety injection phase and close with initiation of recirculation.

REFERENCE: (1) Report No FaAA-84-12-14 (Revision 1.0), Evaluation of Transient Conditions on Emergency Diesel Generator Crankshafts at San Onofre Nuclear Generating Station, Unit 1.

3.7.2 ELECTRICAL SUPPLY: SHUTDOWN

APPLICABILITY: MODES 5 and 6

OBJECTIVE: To define those conditions of available electrical power to ensure that the station can be maintained in the shutdown or refueling condition for extended period .

SPECIFICATION: a. One Southern California Edison Company or San Diego Gas and Electric Company high voltage transmission line to the switchyard and one transmission circuit from the switchyard, immediate or delayed access, to the onsite safety-related distribution system shall be OPERABLE.

b. One diesel generator shall be OPERABLE which is capable of automatic start, with a total connected design load not to exceed 6,000 kW and with:

1. A day tank containing a minimum 290 gallons of fuel,

2. A fuel storage system containing a minimum of 37,500 gallons of fuel, and

3. A fuel transfer pump.

c. One train of AC buses sh 'l be OPERABLE comprised of:

1. 4,160 volt Bus 1C, and 480 volt Buses 1 and 3 with at least one associated station service transformer; OR

2. 4,160 volt Bus 2C, and 480 volt Buses 2 and 4 with at least one associated station service transformer.

d. 120 volt Vital Buses 1, 2, and 4 energized from associated inverters connected to DC Bus 1.

e. One 125 volt DC Bus OPERABLE and energized from the associated battery with at least one full capacity charger.

ACTION: With less than the minimum required AC and DC electrical sources specified above, suspend all operations involving CORE ALTERATIONS or positive reactivity changes. Initiate corrective actions to energize the required electrical buses. Within 8 hours, depressurize and vent the RCS through at least a 1.75 square inch vent.

BASIS: In MODES 5 and 6, the requirement for one source of offsite power and one diesel generator to be OPERABLE will provide diverse and redundant electrical power sources in order that the station can be maintained in the COLD SHUTDOWN or REFUELING condition for extended time periods. Additionally, this

3.8 FUEL LOADING AND REFUELING

APPLICABILITY: Applies to fuel handling and refueling operations. For the applicable surveillance requirements, see Table 4.1.2.

OBJECTIVE: To prevent incidents during fuel handling operations that could affect public health and safety.

SPECIFICATIONS: A. During refueling operations (MODE 6):

1. Radiation levels in the containment and spent fuel building shall be monitored.

2. Core subcritical neutron flux shall be continuously monitored during the entire refueling period by not less than two neutron monitors, each with continuous visual indication and one with continuous audible indication.

3. For water levels in the refueling pool greater than elevation 40 feet, 3 inches (See 7. below for reference evaluation), the following specifications shall apply:

a. At least one of the following methods of decay heat removal shall be in operation and circulating reactor coolant at a flow rate of ≥ 400 gpm:

(1) One RHR TRAIN.

(2) One refueling water pump taking suction from the refueling pool through the recirculation heat exchanger (with supporting heat removal systems operating), and discharging via the safety injection system piping to one reactor coolant loop cold leg.

b. With less than one method of decay heat removal in operation, except as provided in c. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system. Immediately initiate corrective action to return the required decay heat removal method to operating status as soon as possible. In addition, within four hours, close all containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere.

c. The decay heat removal capability may be removed from operation for up to one hour per eight hour period.

6. The reactor shall be subcritical for at least 148 hours prior to movement of irradiated fuel in the reactor pressure vessel.

7. Borated water to ensure the SHUTDOWN MARGIN as specified in Item A.(5) above shall be maintained to an elevation not less than 40 feet 3 inches in the refueling pool during movement of fuel assemblies and RCC's. Reference elevation is sea level, mean lower low water.

8. If any of the specified limiting conditions for refueling is not met, refueling of the reactor shall cease, work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be carried out.

B. With fuel assemblies in the spent fuel storage pool:

1. Loads in excess of 1,500 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

2. Borated water to ensure the SHUTDOWN MARGIN as specified in Item A(5) above shall be maintained to an elevation not less than 40 feet 3 inches in the spent fuel storage pool. Reference elevation is sea level, mean lower low water.

3. With the requirement of B(2) above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limits within four hours.

<u>BASIS</u>: During refueling the reactor cavity is filled with approximately 240,000 gallons of borated water whose concentration is sufficient to maintain the reactor subcritical by greater than 5% $\Delta k/k$ or to a boron concentration greater than or equal to 2,000 ppm, whichever is more restrictive. Operation of one method of decay heat removal is provided to assure continuous mixing flow of refueling water through the reactor vessel during the refueling period.[1] Borated water injection capability is provided as per Specification 3.2 Part A in the unlikely event there is any need during the refueling period.

3.9 MODERATOR TEMPERATURE COEFFICIENT (MTC)

APPLICABILITY: Applies to negative moderator temperature coefficient (MTC) during core operations whenever the nominal reactor coolant inlet temperature is greater than or equal to 528·F.

OBJECTIVE: To establish negative MTC limits for the core.

SPECIFICATION: a. The MTC shall be less negative than - 3.8 x $10^{-4}$ Δk/k/·F for all rods withdrawn, end of cycle life (EOL) and the RATED THERMAL POWER condition.

b. In order to assure that the above negative MTC limit is not exceeded, the MTC shall be measured at any THERMAL POWER and compared to the predetermined, calculated negative MTC within 7 effective full power days (EFPD) of reaching an equilibrium boron concentration of 300 ppm. The predetermined calculated MTC value (at RATED THERMAL POWER conditions with all rods withdrawn) is determined as follows:

-3.1 x $10^{-4}$ Δk/k/·F MTC at a nominal core inlet coolant temperature of 551.5·F, and MTC increasing linearly with decreasing inlet coolant temperature to -2.5 x $10^{-4}$ Δk/k/·F at a nominal core inlet coolant temperature of 528.0·F.

ACTION: In the event this comparison indicates the MTC is more negative than the applicable value given above, the MTC shall be remeasured, and compared to the EOL MTC limit of -3.8 x $10^{-4}$ Δk/k/·F at least once per 14 EFPD during the remainder of the cycle. If the measured MTC is more negative than the -3.8 x $10^{-4}$ Δk/k/·F limit any time during the remainder of the cycle, the reactor shall be in HOT SHUTDOWN within 12 hours after exceeding the negative MTC limit.

BASIS: The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the San Onofre Unit 1 accident and transient analyses.

The limiting MTC used in the steam line break accident analysis is given as a function of $k_{eff}$ and average moderator temperature in Figure 14 of Amendment 18 to the FSA. In order to ensure that the safety analysis remains valid, the reactor should not be operated with an MTC more negative than the limit implied by Figure 14 of Amendment 18.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

BASIS: The percent full power axial offset limits are conservatively established considering the core design peaking factors, 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 IAO to the axial offset monitoring systems(1). The axial offset limit established from the incore versus excore data have been reduced by an amount equivalent to FCC to allow for burnup and time dependent differences between the periodic correlation verification and the monthly correlation check. Correcting the allowed IAO limits by an amount equal to FCC maintains plant operation within the original safety analysis assumptions. Should a specific cycle analysis establish that the analytical determination of the relationship between core peaking factors and IAO has changed in a manner warranting modification to the existing envelope of peaking factor (1,2), then a change to the functional relationship of the specification shall be submitted to the Commission. The incore-excore data correlation is checked or verified periodically as delineated in Specification 3.10, INCORE INSTRUMENTATION.

Reducing power until IAO is within the specified limits in cases when limits are exceeded, will assure that design limits which were set in consideration of accident conditions are not exceeded. In the event that no method exists for determining IAO, actions are specified to place the plant in MODE 2 within 6 hours. 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) Reload Safety Evaluation, San Onofre Nuclear Generating Station, Unit 1, Cycle 10, edited by J. Skaritka, Revision 1, Westinghouse, March, 1989

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

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

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

3.12 CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM

APPLICABILITY: Applies to the operational status of the control room emergency air treatment system.

OBJECTIVE: To identify those conditions of the control room emergency air treatment system which will ensure reliable and efficient operation, should the system be needed.

SPECIFICATION: Effective upon completion of field testing to the modified filter system.

A. Except as specified in Specification 3.12.B below, the control room emergency air treatment system shall be OPERABLE whenever the reactor is to be made or maintained critical. The system will be considered OPERABLE as long as the tests and analyses specified in Specification 4.11 are satisfactorily completed at the required intervals and the system is not removed from service.

B. From and after the date that the control room air treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days.

C. If the conditions in 3.12.B cannot be met, reactor shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN condition within 24 hours.

BASIS: The control room emergency air treatment system is designed to filter the control room intake air during control room isolation. The system is placed in operation under administrative control when conditions warrant its use.

The system utilizes a fan, a high efficiency particulate absolute (HEPA) filter, pre-filters and a charcoal adsorber bed. The pre-filters are installed before the charcoal bed to prevent clogging of the iodine adsorbers. The charcoal adsorbers reduce the potential intake of radioiodine to the control room.

The OPERABILITY requirements of this Specification in conjunction with the surveillance requirement of Specification 4.11 provide reasonable assurance that the system will operate, if needed, at a degree of efficiency equal to or better than that assumed in the Final Safety Analysis.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited time while repairs are made. If the system cannot be repaired within seven days, the reactor is shut down and brought to COLD SHUTDOWN within 24 hours.

## 3.13 SHOCK SUPPRESSORS (SNUBBERS) OPERABILITY

APPLICABILITY: Applies to safety related shock suppressors (snubbers). |

OBJECTIVE: To define operability requirements of snubbers required to protect safety related piping from unrestricted motion when subjected to dynamic loading as might occur during a seismic event or severe transient.

SPECIFICATION:

A. During MODES 1, 2, 3, and 4 (MODES 5 and 6 for snubbers located in the systems required OPERABLE in those MODES) all snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

B. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.14.C on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

BASIS: Snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by in-service functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at refueling outage intervals. Observed failures of these sample snubbers will require functional testing of additional units. Snubbers of rated capacity greater than 120,000 pounds are exempt from functional testing requirements because of the impracticability of testing such large units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc....). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

## 3.14 FIRE PROTECTION

### 3.14.1 FIRE SUPPRESSION WATER SYSTEM

APPLICABILITY: At all times.

SPECIFICATION: The fire suppression water system shall be OPERABLE with:

a. Three fire suppression pumps, each with a rated capacity of at least 1000 gpm, with their discharge aligned to the fire suppression header. One pump must be from Unit 1; the remaining two pumps may be selected from the four pumps available at San Onofre Units 1, 2, and 3.

b. Two separate water supplies (one from Unit 1 and one from Units 2 & 3), each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path from each required water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the first valve upstream of the water flow alarm device on each sprinkler or hose standpipe, and the first valve upstream of the deluge valve on each deluge or spray system required to be OPERABLE per Specifications 3.14.2, 3.14.3, and 3.14.5.

ACTION:

A. With one required pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

B. With the fire suppression water system otherwise inoperable, establish a backup fire suppression system within 24 hours.

BASIS: The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, spray, and/or sprinklers, Halon, and fire hose stations. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

The requirement for OPERABILITY of separate fire water supplies from Unit 1 and Units 2 and 3 (including supply, pumping capability, and piping) will assure two separate supply connections to the Unit 1 fire main loop in accordance with BTP 9.5-1, Appendix A (1976), Section E.2.c. In the event that portions of the fire suppression water system are inoperable, backup fire fighting equipment are required to be made

TABLE 3.14.2.1

REQUIRED SPRINKLER AND SPRAY SYSTEMS

| Fire Area/ Zone | Hazard | Location | System Type |
|---|---|---|---|
| 1 | Reactor coolant pumps, RHR pumps, cable | Inner Containment Sphere | Deluge - borated water spray* |
| | Cable insulation outside secondary shield | Outer Containment Sphere | Deluge - borated water spray* |
| 2A | Charging Pumps | Charging Pump Room | Wet Pipe |
| 4B/4D | Cable Insulation | Cable Trays, Yard/Breezeway Area | Deluge water spray |
| 4D | Transformer oil | Station Service<br>Transformer 1<br>Transformers 2 & 4 | <br>Deluge water spray<br>Deluge water spray |
| 9A | Turbine lubricating oil and cable insulation | System #1 chemical treatment area | Deluge water spray |
| | Turbine lubricating oil and cable insulation | System #2 lube oil reservoir area (north half) | Deluge water spray |
| | Turbine lubricating oil and cable insulation | System #3 lube oil reservoir area (south half) | Deluge water spray |
| | Turbine lubricating oil | System #4 480 V room wall & turbine building north wall | Wet pipe |
| | Turbine lubricating oil and cable insulation | System #5 north turbine building area protection | Wet pipe |
| | Hydrogen seal oil | Hydrogen seal oil unit | Deluge water spray |
| 17A | Diesel Generator | North Diesel Generator | Pre-Action Sprinkler |
| 18 | Diesel Generator | South Diesel Generator | Pre-Action Sprinkler |

*This includes a refueling water pump, 240,000 gallons of borated water in the refueling water storage tank and associated system valves.

## 3.14.5 FIRE HOSE STATIONS

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

SPECIFICATION: The following fire hose stations shall be OPERABLE:

a. See Table 3.14.5.1

ACTION:

A. With one or more of the fire hose stations shown in Table 3.14.5.1 inoperable, route a fire hose* to provide equivalent nozzle flow capacity to the unprotected area(s) from an OPERABLE hose station or alternate fire water supply, within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to plant workers, plant equipment, or the hose itself, a fire hose shall be stored in an area easily accessible to the unprotected area. Signs identifying the purpose and location of the fire hose shall be mounted at the inoperable hose station.

B. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASIS: In the event that a fire hose station is inoperable, the establishment of backup suppression in the affected areas is required to provide fire suppression capability until the inoperable system is restored to operability.

REFERENCES:

1. Fire Protection Program Review, BTP APCSB 9.5-1, San Onofre Nuclear Generating Station, Unit 1, March 1977; submitted to the NRC by letter dated March 16, 1977 in Docket No. 50-206.

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* Fire hose will be run within 1 hour of entering the ACTION statement if an operable water supply is not available within 250 feet of the area protected by the inoperable hose station, or two 150 feet hose packs (1-3/4") on the fire truck are not operable. Fire hose will be supplied by the fire department responding to a fire if an operable water supply is available within 250 feet of the area protected by the inoperable hose station. With the required hose station inside containment inoperable and containment integrity established, fire hose will be supplied only to the nearest access point.

TABLE 3.14.5.1

FIRE HOSE STATIONS

| Fire Area/ Zone | Location | Elevation | Hose Station Number |
|---|---|---|---|
| 1 | Inside Sphere | 42′ - 0" | 25 |
| 2A | Reactor Auxiliary Building, Lower Level | 5′ - 0" | 17 |
| 2P | Boric Acid Injection Pump Room | 20′ - 0" | 16 |
| 4D | Turbine Plant Cooling Water Area | 14′ - 0" | 8 |
| 9A | Chemical Feed and Lubrication-Oil Reservoir Area | 14′ - 0" | 22 |
| 9A | East Feedwater Pump/Condenser Area | 8′ - 6" | 5 |
| 9A | East Feedwater Pump/Condenser Area | 14′ - 0" | 3 |
| 9A | East Feedwater Pump/Condenser Area | 14′ - 0" | 4 |
| 9A | West Feedwater Pump/Condenser Area | 14′ - 0" | 7 |
| 9A | West Feedwater Pump/Condenser Area | 8′ - 6" | 6 |
| 9B | Turbine and Heater Decks | 35′ - 6" | 10 |
| 9B | Turbine and Heater Decks | 35′ - 6" | 11 |
| 9B | Turbine and Heater Decks | 42′ - 0" | 12 |
| 9B | Turbine and Heater Decks | 42′ - 0" | 13 |
| 9B | Turbine and Heater Decks | 35′ - 6" | 14 |
| 9B | Turbine and Heater Decks | 35′ - 6" | 15 |
| 12 | Administration/Control Building First Floor Single-Story Office Area | 20′ - 0" | 26 |
| 11A | Administration/Control Building First Floor Health Physics and Locker Area | 20′ - 0" | 27 |
| 2D | Cryogenic Building Back Yard Area | 20′ - 0" | 24 |

3.14.6 FIRE DETECTION INSTRUMENTATION

APPLICABILITY: Whenever equipment protected by the fire detection instrumentation is required to be OPERABLE.

SPECIFICATION: As a minimum, the following fire detection instrumentation shall be OPERABLE:

a. See Table 3.14.6.1

ACTION:

A. With one or more of the required fire detection instruments inoperable:

1. For areas* outside containment establish within 1 hour an hourly fire watch patrol.

2. For areas* inside containment, establish a fire watch patrol in containment, at least once per 8 hours or monitor the temperature at least once per hour at the locations listed below:

   a. Inside secondary shield:
      A minimum of 2 out of 3 for each RCP: RCP lower water bearing, RCP upper guide bearing, and RCP down thrust bearing;

      AND,

      A minimum of 2 out of 3 of: After RCP motor cooling fan unit, After RCP standby motor cooling fan unit, and RCP motor space

   b. Outside secondary shield:
      A minimum of 3 out of 5 of: Control rod cooler discharge, Control rod shroud air inlet, Reactor cavity air outlet, Sphere space, and Control rod cooler inlet

3. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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* Not required for areas that pose temporary radiation and/or life threatening safety hazards. If the fire watch patrol cannot be restored within 24 hours, prepare and submit a Special Report to Commission pursuant to Specification 6.9.2 within the next 7 days outlining the action taken, the cause of hazards, and the plans and schedule for restoring the required fire watch/patrol.

TABLE 3.14.6.1

FIRE DETECTION INSTRUMENTS

| Fire Area/Zone | Location | Required Instruments Operable Early Warning | Required Instruments Operable Actuation |
|---|---|---|---|
| 1 | Containment Sphere | | |
| | Inside Secondary Shield | 8 | |
| | Outside Secondary Shield | 20 | |
| 2A | Reactor Auxiliary Bldg. | | |
| | Lower Level | 9 | |
| 4A&B | East and West Penetration Areas | 33 | |
| 4C | Doghouse | 2 | |
| 4B/4D | Cable Trays | | |
| | Yard/Breezeway Area | | 2 |
| 4D | Service Transformer 1 | | 2 |
| | Service Transformer 2 & 4 | | 2 |
| 4G | DSD Diesel Generator Enclosure | 6 | |
| | DSD Switchgear/Battery Room | 3 | |
| 7 | 480V Switchgear Room | | 8 |
| 8 | 4160V Switchgear Room | | 16 |
| 9A | Turbine Building Ground Floor | | |
| | Instrument Air Compressors | 1 | |
| | Exciter and MCC 4 Area | 16 | 2 |
| | Lube Oil Reservoir | 29 | 12** |
| 11A | Health Physics and | | |
| | Locker Room | 5* (4 at present) | |
| 11B | HVAC Equipment Room | 3 | |
| 12 | Offices 1st Floor | | |
| | Power Block | 7 | |

* Upon completion of DCP 3449.01.
**Includes 6 line-type detectors.

any fire having the potential of making the existing normal safe shutdown systems unavailable. The DSD incorporates:

- Remote shutdown capability
- Independent onsite power source
- RCS charging capability
- Auxiliary feedwater flow capability
- Independently powered instrumentation and controls

Similarly, alternate shutdown methods rely on existing plant equipment used in off-normal modes to assure safe shutdown capability in the event of certain fires. Table 3.14.8.1 lists the minimum equipment required to be OPERABLE in order to provide these capabilities. Use of the equipment in Table 3.14.8.1, as it is used to mitigate certain fires, is described in the references.

In the event that one or more components listed in Table 3.14.8.1 is rendered inoperable for more than 7 days, the Technical Specifications permit continued plant operation for up to 60 days, if equivalent shutdown capability is provided. The equivalent shutdown capability provided when ASD/DSD equipment is inoperable depends on the specific equipment involved and, therefore, should be sufficient to assure that the intended shutdown actions can be accomplished, or that fires can be reasonably precluded during that time for which ASD/DSD equipment would otherwise be required, consistent with the ASD/DSD design basis. Temporary procedures or special fire watch patrols established to provide this equivalent capability should be approved by the Plant Manager prior to implementation.

Boron concentration at the PASS Skid will be measured utilizing either potentiometric analysis (auto-titrator) or PASS boronometer. Potentiometric analysis utilizing grab samples and an auto-titrator provides a very accurate and reliable method of determining boron concentration. This technique is utilized by Unit 1 chemists routinely in support of plant operations. Continuous sampling of boron concentration is not required since inadvertent dilution of the primary system as a result of the spurious actuation of equipment is not likely since design basis fires causing the loss of neutron source range monitors have been analyzed to not impact the operability of components which have the capability of supplying unborated water to the RCS make up path. The PASS boronometer remains available.

## 3.15 RADIOACTIVE LIQUID EFFLUENTS

### 3.15.1 LIQUID EFFLUENTS CONCENTRATION

APPLICABILITY: At all times.

OBJECTIVE: Maintain the concentration of radioactive liquid material released from the site below 10 CFR 20 limits.

SPECIFICATION: A. The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to $2 \times 10^{-4}$ mCi/ml.

B. ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay restore the concentration to within the above limits.

BASIS: This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

## 3.16 RADIOACTIVE GASEOUS EFFLUENTS

### 3.16.1 DOSE RATE

APPLICABILITY: At all times.

OBJECTIVE: Maintain the dose rate at the exclusion area boundary from radioactive gaseous effluents within 10 CFR 20 limits.

SPECIFICATION:

A. The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following values:

1. The dose rate limit for noble gases shall be ≤ 500 mrem/year to the total body and ≤ 3000 mrem/year to the skin, and

2. The dose rate limit for I-131, I-133, for tritium and for all radionuclides in particulate form with half lives greater than 8 days shall be ≤ 1500 mrem/year to any organ.

B. ACTION:

With the dose rate(s) exceeding the above limits, without delay restore the release rate to within the above limit(s).

BASIS: This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrating of 10 CFR Part 20, Appendix B, Table 11, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to ≤ 1500 mrem/year.

TABLE 3.18.1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

| Exposure Pathway and/or Sample | Number of Samples and Sample Locations[a] | Sampling and Collection Frequency[a] | Type and Frequency of Analyses |
|---|---|---|---|
| 1. AIRBORNE Radioiodine and Particulates | Samples from at least 5 locations<br>3 samples from offsite locations (in different sectors) of the highest calculated annual average ground level D/Q.<br><br>1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q.<br><br>1 sample from a control location 15-30 km (10-20 miles) distant and in the least prevalent wind direction.[c] | Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days.[d] | Radioiodine cartridge. Analysis at least once per 7 days for I-131. Particulate sampler. Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic[b] analysis on each sample When gross beta activity is ≥ 10 times the yearly mean of control samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days. |
| 2. DIRECT RADIATION[e] | At least 30 locations including an inner ring of stations in the general area of the SITE BOUNDARY and an outer ring approximately in the 4 to 5 mile range from the site with a station in each sector of each ring. The balance of the stations are in special interest areas such as population centers, nearby residences, schools, and in 2 or 3 areas to serve as control stations. | At least once per 92 days. | Gamma dose. At least once per 92 days. |

| Exposure Pathway and/or Sample | Number of Samples and Sample Locations* | Sampling and Collection Frequency* | Type and Frequency of Analyses |
|---|---|---|---|
| 3. WATERBORNE | | | |
| a. Ocean | 4 Locations | At least once per month and composited quarterly | Gamma isotopic analysis of each monthly sample. Tritium analysis of composite sample at least once per 92 days. |
| b. Drinking | 2 Locations | Monthly at each location. | Gamma isotopic and tritium analyses of each sample. |
| c. Sediment | 4 Locations from Shoreline | At least once per 184 days. | Gamma isotopic analysis of each sample. |
| d. Ocean | 5 Locations Bottom Sediments | At least once per 184 days. | Gamma isotopic analysis of each sample. |
| 4. INGESTION | | | |
| a. Nonmigratory Marine Animals | 3 Locations | One sample from each group (listed below) will be collected in season, or at least once per 184 days if not seasonal. Groups to be sampled:<br>1. Fish-2 adult species such as flatfish, bass, perch or sheepshead.<br>2. Crustacean-such as crab or lobster.<br>3. Mollusks-such as limpets, clams or seahares. | Gamma isotopic analysis on edible portions. |

3.19 SOLID RADIOACTIVE WASTE

APPLICABILITY: At all times.

OBJECTIVE: Ensure meeting the requirements for the SOLIDIFICATION and shipment of solid radwaste.

SPECIFICATION: A. The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

B. ACTION:

1. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

2. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

BASIS: This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/ solidification/agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3.20 OVERPRESSURE PROTECTION SYSTEMS

APPLICABILITY: Applies to operability of the overpressurization protection systems.

OBJECTIVE: To preclude the potential for exceeding the limits set in 10 CFR 50, Appendix G, in the event of a pressure transient while water-solid.

SPECIFICATION: A. When the RCS pressure is ≤ 400 psig* and pressurizer water level is greater than 50%, at least one of the following overpressure protection systems shall be OPERABLE:

(1) Two power operated relief valves (PORVs) with a lift setting of ≤ 500 psig,** or

(2) A reactor coolant system vent(s) of ≥ 1.75 square inches.

ACTION: B. With one PORV inoperable when required in accordance with Specification A above, either restore the inoperable PORV to OPERABLE status within seven days or depressurize and vent the RCS through a 1.75 square inch vent(s) within the next eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

C. With both PORVs inoperable when required in accordance with Specification A above, depressurize and vent the RCS through at least a 1.75 square inch vent(s) within eight hours; maintain the RCS in a vented and tagged condition until both PORVs have been restored to OPERABLE status.

D. In the event either the PORVs or the RCS vent(s) are used to mitigate a potential RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances indicating transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

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* The placing in service of the OMS at ≤ 400 psig is intended to assure that protection is provided whenever temperature is below 360°F. The alarm to arm the OMS being keyed to pressure assures that inadvertent opening of the PORVs does not occur due to placing the OMS into service with RCS pressure above the 500 psig initiation setpoint.

** The 500 psig setpoint is based on the current heatup and cooldown curves for 16 EFPY. The setpoint requires reevaluation for acceptability any time the curves are changed.

## SURVEILLANCE REQUIREMENTS (GENERAL)

APPLICABILITY: Applies to the surveillance requirements to be implemented in these specifications.

OBJECTIVE: To define the conditions under which the surveillance requirements of Section 4 Specifications are applicable.

SPECIFICATION:

4.0.1 Surveillance Requirements shall be met during OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

BASIS:

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational

APPLICABILITY: Applies to the surveillance requirements to be implemented in these specifications.

OBJECTIVE: To define the conditions under which the surveillance requirements of Section 4 Specifications are applicable.

SPECIFICATION:

4.0.1 Surveillance Requirements shall be met during OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25% of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements.

BASIS:

Specifications 4.0.1 through 4.0.4 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational

status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refuelling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by

# 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)<br>R (3,4) | M | N.A. | N.A. |
| 3. Power Range, Neutron Flux, Dropped Rod Stop | N.A. | N.A. | M | N.A. | N.A. |
| 4. Intermediate Range, Neutron Flux | S | R (3,4) | S/U (1),<br>M | N.A. | N.A. |
| 5. Source Range, Neutron Flux | S | R (3) | S/U (1),<br>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. |
| 13. Reactor Coolant Pump Breaker Position* | S | R | R | N.A. | N.A. |
| 14. 4kV Bus 1C and Bus 2C Voltage | N.A. | R | R | N.A. | N.A. |

*Applicable to Item 7 in Table 2.1

## TABLE 4.1.2 (continued)

| | | Check | Frequency |
|---|---|---|---|
| 2. | Safety Injection Line and RWST 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 |
| 3. | Control Rod Drop | a. Verify that all rods move from full out to full in, in less than 2.44 seconds | At each refueling shutdown |
| 4. | (Deleted) | | |
| 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 |
| 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. | Not used. | | |

TABLE 4.1.2 (continued)

| | | Check | Frequency |
|---|---|---|---|
| 11. | (Deleted) | | |
| 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. | Monthly, when the reactor is critical, and prior to returning reactor to critical, when period of subcriticality extended the test interval beyond one month. |
| 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. | Monthly, when the reactor is critical and prior to returning reactor to critical, when period of subcriticality extended the test interval beyond one month. |
| 14a. | Spent Fuel Pool Water Level | Verify water level per Technical Specification 3.8. | a. Once every seven days when spent fuel is being store 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. |
| 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 and in MODE 3 with reactor trip breakers closed, verify that all required reactor coolant loops are in operation and circulating reactor coolant. | a. once per 12 hours. |
| | | b. Per Technical Specification 3.1.2.E, in MODE 3 with the reactor trip breakers open, verify. | |

4.1.6 PRESSURIZER RELIEF VALVES

APPLICABILITY: Applies to the power operated relief valves (PORVs) and their associated block valves for MODES 1, 2 and 3.

OBJECTIVE: To ensure the reliability of the PORVs and block valves.

SPECIFICATION:

A. Each PORV shall be demonstrated OPERABLE:

1. At least once per 31 days by performance of a CHANNEL TEST, which may include valve operation, and

2. At least once per 18 months by performance of a CHANNEL CALIBRATION.

B. Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel, unless the block valve is being maintained closed in order to meet the requirements of Specification 3.1.5.A.

C. The backup nitrogen supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by transferring motive power from the normal air supply to the nitrogen supply and operating the valves through a complete cycle of full travel.

BASIS: The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The air supply for both the relief valves and the block valves is capable of being supplied from a backup passive nitrogen source to ensure the ability to seal this possible RCS leakage path.

REFERENCES:

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

## 4.2 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM

### 4.2.1 SAFETY INJECTION AND CONTAINMENT SPRAY SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Safety Injection System and the Containment Spray System.

OBJECTIVE: To verify that the Safety Injection System and the Containment Spray System will respond promptly and proporly if required.

SPECIFICATION: I. System Tests

A. Hot Safety Injection System Test

(1) When the plant is planned to be shutdown from MODE 1 operation and is planned to enter MODE 5 operation, a Hot SIS Test shall be performed in MODE 3 while RCS pressure is above 1500 psi but not more often than once every 9 months. The test shall include a determination of the force required to open valves HV-851 A and B and the margin of available actuation force.

(2) The test will be considered satisfactory if:

(a) control board indication and visual observations indicate all components have operated and sequenced properly. That is, the appropriate pumps have started and/or stopped and started, and all valves have completed their travel.

(b) the measured actuator force for both the HV-851 A and B valves is equal to or less than 10,000 lb,.*

(3) If the measured actuator force of either HV-851 A or B is between 10,000 and 22,000 lb,, the HV-851 A and B valves shall be considered OPERABLE but the future testing interval shall be accelerated as determined by the following equation:

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*Upon receipt of satisfactory data from continuing testing and analysis, the NRC staff will consider a request from Southern California Edison Company to change this number to more accurately reflect existing conditions.

$$T = T_L \frac{(22{,}000 - F)}{12{,}000}$$

where: T = maximum time in days of operation allowed before next surveillance test is required

$T_L$ = time in days of operation since the last surveillance test

F = measured actuator force

(4) If the measured actuator force of either HV-851 A or B is greater than 22,000 lb., test results shall be reported to the NRC pursuant to Specification 6.9.2 along with proposed corrective actions. NRC approval shall be obtained prior to returning the unit to service.

B. Trisodium Phosphate Test

(1) A test of the trisodium phosphate additive shall be conducted once every refueling to demonstrate the availability of the system. The test shall be performed in accordance with the following procedure:

(a) The three (3) storage racks are visually observed to have maintained their integrity.

(b) The three (3) racks, each with a storage capacity of 1800 pounds of anhydrous trisodium phosphate additive, are visually observed to be full.

(c) Trisodium phosphate from one of the sample storage racks inside containment shall be submerged without agitation in 25±0.5 gallons of 150°F to 175°F distilled water borated to 3900±100 ppm boron.

(2) The test shall be considered satisfactory if the racks have maintained their integrity, the racks are visually observed to be full, and the trisodium phosphate dissolves to the extent that a minimum pH of 7.0 is reached within 4 hours of the start of the test.

B. Leakage Testing

(1) The recirculation loop outside containment (including the Containment Spray System) shall be pressurized at a pressure equal to or greater than the operating pressure under accident conditions at intervals not to exceed the normal plant refueling interval. Visual inspections for leakage shall be made and if leakage can be detected, measurements of such leakage shall be made. In addition, pumps and valves of the recirculation loop outside containment which are used during normal operation, shall be visually inspected for leakage at intervals not to exceed once every six months. If leakage can be detected, measurements of such leakage shall be made.

(2) The non-redundant Containment Spray System piping shall be visually inspected at intervals not to exceed the normal plant refueling interval. Observations made as part of compliance with Paragraph C, above, or Paragraph I.C(2) of Technical Specification 4.2 will be acceptable as visual inspection of portions of non-redundant Containment Spray System piping.

C. RWST Low Level Trips

Monthly, perform a CHANNEL TEST and every refueling interval, perform a CHANNEL CALIBRATION, of the SI/Feedwater Pump trip and the MOV 850A, 850B and 850C automatic closure on low-low Refueling Water Storage Tank level.

<u>BASIS</u>: The Safety Injection System is a principal plant safeguard. It provides means to insert negative reactivity and limits core damage in the event of a loss of coolant or steam break accident.

Preoperational performance tests of the components are performed in the manufacturer's shop. An initial system flow test demonstrates proper dynamic functioning of the system. Thereafter, periodic tests demonstrate that all components are functioning properly. For these tests, flow through the system is generally not required. However, in the case of the "Hot SIS Test," actual conditions of an SI event are simulated. This test is performed to assure that long-term set of the valve seat faces on HV-851 A and B has not caused the valves to become inoperable. The test is required to be performed as the plant is shutting down from MODE 1 in order to assure that the valves have not been disturbed (i.e., the long-term set is still in effect) and that full dynamic conditions that would occur during an actual SI event are simulated. When possible, the test should be performed prior to stopping the feedwater pumps (this is not a requirement). This will further assure that the valves will be in the same condition as when required for an actual Safety Injection event since the discharge pressure of the feedwater pumps acting on the valves will keep them seated even considering any backpressure built up in the downstream SI header. The equation used to determine future intervals if actuator force is between 10,000 lb, and 22,000 lb, is developed by shortening the interval in direct proportion to the degree to which the force exceeds 10,000 lb,. During the test, all components are verified to have operated and sequenced properly.

The tests required in this specification will demonstrate that all components which do not normally and routinely operate will operate properly and in sequence if required. The portion of the Recirculation system outside the containment sphere is effectively an extension of the boundary of the containment. The measurement of the recirculation loop leakage ensures that the calculated EAB 0-2 hr. thyroid dose does not exceed 10 CFR 100 limits.

The trisodium phosphate stored in storage racks located in the containment is provided to minimize the possibility of stress corrosion cracking of metal components during operation of the ECCS following a LOCA. The trisodium phosphate provides this protection by dissolving in the sump water and causing the final pH to be raised to 7.0 - 7.5. The requirement to dissolve trisodium phosphate from one of the sample storage racks in distilled water heated and borated to the extent recirculating post LOCA sump water is projected to be heated and borated, provides assurance that the stored trisodium phosphate will dissolve as required following a LOCA. The sample storage racks are sized to contain 0.5 pounds of trisodium phosphate. Trisodium phosphate stored in the sample storage racks has a surface area to volume ratio of 1.33 whereas the trisodium phosphate stored in the main racks has a surface area to volume ratio of 1.15.

Visual inspection of the non-redundant piping in the Containment Spray System provides additional assurance of the integrity of that system.

Surveillance testing of the RWST low-low level main feedwater/ safety injection pump trips and automatic closure of MOV 850A, 850B and 850C valves will ensure that these components will be available to complete their safety functions if required.

## 4.2.2 PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES TESTING (SURVEILLANCE REQUIREMENT)

APPLICABILITY: Applies to the operational status of the primary coolant system pressure isolation valves during MODES 1, 2 and 3.

OBJECTIVE: To increase the reliability of primary coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

SPECIFICATION: 1. Periodic leakage testing(a) on each valve listed in Table 3.3.5-1 shall be accomplished every time the plant is placed in the cold shutdown condition for refueling, each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.

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(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. The minimum test differential pressure shall not be less than 150 psid.

B. <u>Acceptance Criteria</u>

Visual inspection for leakage shall be made and if leakage can be detected, measurements of such leakage shall be made. The maximum effective leakage shall be maintained in accordance with Section 3.3.1.A(4) of Appendix A Technical Specifications.

C. <u>Test Schedule</u>

Visual inspections of the recirculation loop outside containment (including the Containment Spray System) shall be made at intervals not to exceed the normal plant refueling interval. In addition, pumps and valves of the recirculation loop outside containment which are used during normal operation, shall be visually inspected for leakage at intervals not to exceed once every six months.

V. <u>Test Result Report</u>

The results of Type A, B, and C leakage rate tests are submitted to the NRC in a summary technical report approximately three months after the conduct of the Type A tests. This report contains an analysis and interpretation of the Type A test results and a summary of periodic Type B and C tests performed since the last Type A test. Leakage rate test results from Type A tests that fail to meet the acceptance criteria specified in Section I.B above are reported in a separate attached summary report that includes an analysis of the test data, an instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to failure in meeting the acceptance criteria. Results and analysis of the supplemental verification test used to demonstrate the validity of the Type A test measurements are included.

VI. <u>Containment Modification</u>

Any major modification or replacement of a component that is part of the containment boundary is followed by Type A, B, or C tests as applicable. The results of such tests are included in the test result report described above and meet the respective acceptance criteria. Minor modifications or replacements performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

## 4.3.2 CONTAINMENT ISOLATION VALVES

APPLICABILITY: Applies to the containment isolation valves listed in Table 3.6.2-1 for MODES 1, 2, 3 and 4.

OBJECTIVE: To ensure reliability of containment isolation valves.

SPECIFICATION:

A. The isolation valves specified in Table 3.6.2-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test.

B. Each isolation valve specified in Table 3.6.2-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

1. Verifying that on a containment isolation test signal, each automatic isolation valve actuates to its isolation position.

2. Verifying that on a containment radiation-high test signal, each purge supply and purge outlet automatic valve actuates to its isolation position.

BASIS: The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

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

4.3.3 HYDROGEN MONITORS AND HYDROGEN RECOMBINERS

APPLICATION: MODES 1 and 2.

OBJECTIVE: To ensure reliability of the hydrogen monitors and hydrogen recombiners required for the detection and control of hydrogen gas.

SPECIFICATION:

A. Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

1. One volume percent hydrogen, balance nitrogen.

2. Four volume percent hydrogen, balance nitrogen.

B. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per year by verifying that a heater sheath temperature of at least 1225 ± 10·F can be attained.

C. Each hydrogen recombiner system shall be demonstrated OPERABLE at least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits.

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits or foreign materials, etc.), and

3. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the test in Specification B above. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

BASIS: The OPERABILITY of the equipment and systems required for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with radiolytic decomposition of water and corrosion of metals within containment.

4.4 EMERGENCY POWER SYSTEM PERIODIC TESTING

APPLICABILITY: Applies to testing of the Emergency Power System.

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.

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 performs a DG SLOW START[1] from standby conditions,

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

c. Verifying the diesel generator is synchronized and running at 6,000 kW (+100 kW, -500 kW) for ≥ 60 minutes,

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

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.

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[1]All diesel starts for testing and surveillance will be slow starts (greater than 24 seconds duration) except for the fast start required by Technical Specification 4.4.F conducted once per 18 months during shutdown and any other fast start required following specific maintenance involving the fast start capability.

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 without automatic SIS/SISLOP tripping circuitry open between redundant buses at least once per 7 days by verifying correct breaker alignment and power availability.

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

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.

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:

      (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 $150 \times 10^{-6}$ ohms, and

      (3) The average electrolyte temperature of ten connected cells is above 61°F for battery banks associated with DC Bus No. 1 and DC Bus No. 2 and above 48°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 anticorrosion material,

(3) The resistance of each cell-to-cell and terminal connection is less than or equal to $150 \times 10^{-6}$ ohms,

(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 invertor.

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 Sequencer (SLSS) 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 Sequencer shall be demonstrated OPERABLE at least once per 18 months during shutdown by:

1. Simulating SISLOP*, and:

   a. Verifying operation of circuitry which locks out non-critical equipment,

   b. Verifying the diesel performs a DG FAST START 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,

   c. Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed.

2. Verifying the generator capability to reject a load of 4,000 kW without tripping. The generator voltage shall not exceed 4,800 volts and the generator speed shall not exceed 500 rpm (nominal speed plus 75% of the difference between nominal speed and the overspeed trip setpoint) during and following the load rejection.

G. Manual Transfer Switches

1. Verify once every 31 days that the fuse block for breaker 8-1181 in MCC-1 for MTS-7 is removed.

2. Verify once every 31 days that MTS-8 is energized from breaker 8-1480B from MCC-4 and the cabinet door is locked, and that breaker 8-1122 from MCC-1 is locked open.

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* SISLOP is the signal generated by a sequencer on coincident loss of voltage on its associated 4160 volt bus (Bus 1C or 2C) and demand for safety injection.

** The sum of all loads on the engine shall not exceed 6,000 kW.

H. Periodic maintenance, surveillance, overhaul and inspection of the required diesel generator shall comply with the following:

1. A diesel engine maintenance and surveillance program as described in the Safety Evaluation related to Amendment No. 123 to this Operating License will be implemented Changes to this program will be subject to the provisions of 10 CFR 50.59.

2. The frequency of major diesel engine overhaul that is a part of the diesel engine maintenance and surveillance program shall be at least once every ten years. For this overhaul, one engine may be inspected during the refueling outage immediately prior to the ten years and the other engine inspected during the refueling outage immediately following the ten years. Alternatively, both inspections may be performed coincident with the 10-year reactor vessel inservice inspection. The 10-year overhaul interval shall be determined on a calendar basis from the date of completion of the last overhaul.

3. Oil hole locations in journals 8 through 12 on each crankshaft shall be inspected with liquid penetrant. This inspection shall be performed at each refueling outage or at the end of fifty[2] start-stop cycles on the engine since the previous inspection, whichever comes first. Indications found shall be evaluated with eddy current testing as appropriate.

   During each major engine overhaul, the fillets of main journal Nos. 4 through 12 should be inspected together with the oil holes, using liquid penetrant. Indications found shall be evaluated with eddy current testing as appropriate. In addition, these inspections should be performed for the oil holes and fillets in at least three of the crankpin journals at each major engine overhaul.

   If during the oil hole and fillet inspections described above, cracks are found in the oil holes or in other crankshaft surfaces, these findings are to be reported to the NRC within 24 hours. The affected engine is to be considered inoperable and is not to be restored to OPERABLE status until the disposition and/or corrective actions have been approved by the NRC staff.

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[2]Start-stop cycles associated with idle (no load) engine operation at 200 rpm or less need not be counted toward the limit of fifty.

4. Cylinder blocks shall be inspected for "ligament" cracks, "stud-to-stud" cracks and "stud-to-end" cracks as defined in the report[3] by Failure Analysis Associates, Inc. (FaAA) entitled "Design Review of TDI R-4 and RV-4 Series Emergency Diesel Generator Cylinder Blocks" (FaAA Report No. FaAA-84-9-11.1) and dated December 1984. (Note that the FaAA report specifies additional inspections to be performed for blocks with "known" or "assumed" ligament cracks.) The inspection intervals (i.e., frequency) shall not exceed the intervals calculated using the cumulative damage index model in the subject FaAA report. In addition, inspection methods shall be consistent with or equivalent to those identified in the subject FaAA report.

   Blocks determined in the future to have "ligament" cracks as the result of the above inspections should be inspected at each refueling outage to determine whether or not cracks have initiated on the top surface, which was exposed because of the removal of two or more cylinder heads. This process should be repeated over several refueling outages until the entire block has been inspected. If after this process has been completed new "ligament" cracks are found, this process should again be repeated. Liquid penetrant testing or a similarly sensitive nondestructive testing technique should be used as appropriate to determine the depth of any cracks discovered.

   Whenever diesel generator No. 1 is operated in excess of 4,375 kW for one hour or more, a visual inspection of the right bank cylinder block is to be performed under intense light within 48 hours after engine shutdown to verify the absence of "stud-to-stud" and "stud-to-end" cracks.

   If "stud-to-stud" or "stud-to-end" cracks are found, these findings are to be reported to the NRC within 24 hours. The affected engine is to be considered inoperable and is not to be restored to OPERABLE status until the disposition and/or corrective actions have been approved by the NRC staff.

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[3]This report was transmitted to H.R. Denton, (NRC), from C.L. Ray, Jr., (TDI Owners Group), by letter dated December 11, 1984.

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 [3] value for each connected cell |
| 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[a] Gravity | ≥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.

4.6 RADIOACTIVE GASEOUS EFFLUENTS

4.6.1 DOSE RATE

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose rate due to the discharge of radioactive gaseous effluents is maintained within 10 CFR 20 limits.

SPECIFICATION:

A. The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of Specification 3.16.1 in accordance with the methods and procedures of the ODCM.

B. The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the limits of Specification 3.16.1 in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.6.1.1.

BASIS: This specification is provided to ensure that the dose rate at and beyond the SITE BOUNDARY from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive materials discharged in gaseous effluents will not result in the exposure of an individual in an UNRESTRICTED AREA, either within or outside the exclusion area boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 [10 CFR 20.106(b)]. For individuals who may at times be within the exclusion area boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the exclusion area boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above "background" to an individual at or beyond the exclusion area boundary to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to ≤ 1500 mrem/year for the nearest cow to the plant.

4.6.3 DOSE, IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

APPLICABILITY: At all times.

OBJECTIVE: To verify the dose due to iodine-131, iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days is maintained as low as is reasonably achievable.

SPECIFICATION: Cumulative does contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the ODCM at least once per 31 days.

BASIS: This specification implements the requirements in Section III.A of Appendix I 10 CFR Part 50, that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated.

The ODCM equations provided for determining the actual doses are based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation and subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

## 4.8 REACTIVITY ANOMALIES

APPLICABILITY: Applies to potential reactivity anomalies.

OBJECTIVE: To require evaluation of reactivity anomalies within the reactor.

SPECIFICATION: A. Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted concentrations reaches the equivalent of one percent in reactivity, an evaluation as to the cause of the discrepancy shall be made within 30 days and reported to the NRC pursuant to Specification 6.9.2.

BASIS: To eliminate expected errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burn-up and reactivity compared with that predicted. This process of normalization should be completed after about 10% of the total core burn-up. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1% would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1% is considered a safe limit since a reactivity insertion of this amount would not result in pressure or temperature transients which exceed the design conditions of the reactor coolant system.

4.10 AUGMENTED INSERVICE INSPECTION OF HIGH ENERGY LINES OUTSIDE CONTAINMENT

APPLICABILITY: Applies to welds in piping systems or portions of systems located outside containment where protection from the consequences of postulated pipe breaks is not provided by a system of pipe whip restraints, protective enclosures, or other measures specifically designed to cope with such breaks.

OBJECTIVE: To provide assurance of the continued integrity of the piping systems over their service lifetime.

SPECIFICATION: A. For the welds in the main steam, main feedwater, and first point extraction lines identified in Reference 1, Table 1 and Table IA, Column: "Break Point Location", for which inservice inspection is specified in Column: "Solution",

(1) At refueling outage No. 4, a baseline inspection consisting of a volumetric examination of all specified welds shall be performed in accordance with the requirements of ASME Section XI Code, "Inservice Inspection of Nuclear Reactor Coolant Systems" 1971, up to and including 1972 addenda.

(2) Subsequent to the baseline examination, the inservice inspection of each weld shall be performed in accordance with the requirements of the Edition and Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" as required by Technical Specification 4.7, with the following schedule:

(a) During the first 3-1/3 years (or nearest refueling outage), volumetric examination of 100% of all welds.

(b) Every 10 years thereafter (or nearest refueling outage), volumetric examination of 33-1/3% of the welds at the expiration of each 1/3 of the inspection interval with a cumulative 100% coverage of all welds every 10 years.

## 4.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

APPLICABILITY: Applies to the leakage of radioactive source materials.

OBJECTIVE: To verify the physical integrity of portable and fixed radioactive calibration sources.

SPECIFICATION:

A. Byproduct material sealed sources which exceed the quantities listed in 10 CFR 30.71, Schedule B, and all other sealed sources containing greater than 0.1 microcuries shall be leak tested in accordance with Specification B, C and D below.

Exception: Not withstanding the periodic leak test required by this specification, any licensed sealed source is exempt from such leak tests when the source contains 100 microcuries or less of beta and/or gamma emitting material or 10 microcuries or less of alpha emitting material.

B. Each sealed source containing radioactive material, other than Hydrogen 3, with a half life greater than thirty days and in any form other than gas, shall be tested for leakage and/or contamination prior to use out of storage and prior to transfer to another person and thereafter at intervals not to exceed six months. This test does not apply to sealed sources that are stored and not in use.

C. The leakage test shall be capable of detecting the presence of .005 microcuries of radioactive material. The test sample shall be taken from the sealed source or from the surfaces of the device in which the sealed source is permanently mounted or stored on which one might expect contamination to accumulate.

D. If testing reveals the presence of .005 microcuries or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or disposed of in accordance with applicable regulatory requirements and reported in the subsequent annual report filed pursuant to Specification 6.9.1.4.

BASIS: This Specification assures that leakage from radioactive material sources does not exceed allowable total body or organ limits. In the unlikely event that those quantities of radioactive byproduct materials of interest to this Specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39 (c) limits for plutonium.

## 4.16 INSERVICE INSPECTION OF STEAM GENERATOR TUBING

APPLICABILITY: Applies to the inservice inspection and sampling selection for steam generator tubing.

OBJECTIVE: To monitor the integrity of the steam generator tube primary boundary and provide guidance for corrective action when imperfections are observed.

SPECIFICATION:

### A. GENERAL STEAM GENERATOR TUBE SELECTION

The steam generators shall be inspected when shutdown by selecting steam generator tubes on the following basis:

1. Tubes for the inspection shall be selected on a random basis except where experience at San Onofre Unit 1 or experience in similar plants indicates critical areas to be inspected.

2. Each inspection shall include at least 3 percent of the total number of tubes in each steam generator to be inspected.

3. Inservice inspections may be limited to one steam generator on a rotating schedule encompassing 3 percent of the total tubes of steam generators in the plant if the results of previous inspections indicate that all steam generators are performing in a like manner.

4. Every inspection shall include all non-plugged tubes, in the steam generator(s) to be inspected that previously had detected imperfections greater than 20 percent.

### B. SUPPLEMENTARY INSPECTIONS

If the inspections in Specification A indicate imperfections, additional inspections shall be required as follows:

1. If any of the tubes inspected pursuant to Specification A.3 are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration since their last inspection, inspect 3 percent of the tubes in one of the uninspected steam generators.

2. If more than 10 percent of the tubes inspected in a steam generator are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration since their last inspection, or one or more of the tubes inspected have an imperfection in excess of the plugging limit, inspect an additional 3 percent of the tubes in that steam generator, concentrating on tubes in those areas of the tube sheet array where tubes with imperfections were found and on that length of tube where the imperfections were found. In addition, the rest of the steam generators shall be inspected in accordance with Specification A.2.

3. If the additional inspection in Specification B.2 indicates that more than 10 percent of the additionally inspected tubes are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration since their last inspection, or one or more of the additionally inspected tubes have an imperfection in excess of the plugging limit, inspect an additional 6 percent of the tubes in that steam generator in the area of the tubesheet array where tubes with imperfections were found and through that length of tube where the imperfections were found.

C. <u>INSPECTION FREQUENCY</u>

The inspections in Specifications A and B above shall be performed at the following frequencies:

1. Inservice inspections shall be not less than 10 nor more than 24 calendar months after the previous inspection.

2. If two consecutive inspections indicate that less than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, the inspections shall be not less than 10 nor more than 40 calendar months after the previous inspection.

3. Unscheduled inspections shall be conducted in accordance with Specification A in the event of primary-to-secondary leaks exceeding Specification 3.1.4 a seismic occurrence greater than an operating basis earthquake, a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

D. <u>ACCEPTANCE CRITERIA</u>

1. As used in this specification:

   a. <u>Imperfection</u> means an exception to the dimensions, finish, or contour required by drawing or specification. Detectable eddy current testing indications below 20 percent of the nominal tube wall thickness, may be considered as imperfections.

   b. <u>Degradation</u> means a service-induced cracking, wastage, pitting, wear or general corrosion occurring on either inside or outside of a tube.

   c. <u>Degraded Tube</u> means a tube containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation above the tube roll expansion. Also, a tube with an imperfection in the region one-half inch below the uppermost one inch of sound roll, is considered a degraded tube.

   d. <u>Defect</u> means an imperfection of such severity that it exceeds the plugging limit or an imperfection located in the uppermost one inch of the tube roll expansion. A tube or sleeve containing a defect is defective.

e. Plugging Limit means the imperfection depth at or beyond which plugging of the tube must be performed. The plugging limit is equal to or greater than 50 percent of the nominal tube wall thickness, except where sleeves are installed, in which case the plugging limit is equal to or greater than 40 percent of the nominal sleeve wall thickness.

For the tube roll expansion region, the following criteria apply:

(i) Any tube containing an imperfection within the uppermost one inch of sound roll shall be considered defective.

(ii) Any imperfection is acceptable below the uppermost one inch of sound roll.

f. Tube Roll Expansion means that portion of the tube which has been increased in diameter by a rolling process.

g. Sound Roll means a tube roll expansion which is fully expanded such that no crevice exists between the outside diameter of the tube and the tubesheet.

2. If, in the inspections performed under Specification A,

a. Less than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, and

b. No tube inspected exceeds the plugging limit,

plant operation may resume.

3. If, in the inspections performed under Specification B,

a. Less than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, and

b. No more than 3 of the tubes inspected exceed the plugging limit,

plant operation may resume after performance of the corrective action in Specification E.

4. If, in the inspections performed under Specification B,

a. More than 10 percent of the total tubes inspected are degraded tubes that were not classified as degraded tubes during the previous inspections or have previously detected imperfections that have increased more than 10 percent wall penetration, or

b. More than 3 of the tubes inspected exceed the plugging limit, prior to resumption of plant operation, the situation shall be reported in a Special Report to the Commission in accordance with Technical Specification 6.9.2 for approval of the proposed remedial action.

5. The results of inspections performed under specifications A or B for all tubes in service which have defects below the uppermost one inch of tube roll expansion shall be reported to the Commission in a Special Report pursuant to Technical Specification 6.9.2 at least seven days prior to the resumption of plant operation. The report shall include:

   a. Identification of applicable tubes, and

   b. Location and size of the degradation.

E. CORRECTIVE ACTION

All leaking tubes, defective tubes, and tubes with imperfections exceeding the plugging limit shall be repaired or plugged.

BASIS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the Reactor Coolant System will be maintained. The program for inservice inspection of steam generator tubes is based on Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = .3 gallons per minute per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of .3 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require shutdown during which the leaking tubes will be located and plugged and additional inspections performed.

If a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 50 percent of the tube nominal wall thickness, except where sleeves are installed, in which case the plugging limit is 40 percent of the nominal sleeve wall thickness. A plugging limit of 50 percent for tubes and 40 percent for sleeves ensures that defects will not occur between inspection intervals. Tubes with defects below the uppermost one inch of a sound roll may remain in service, provided there are no imperfections in this portion of the tube. The distance of one inch includes a 0.25 inch eddy current measurement uncertainty.

The results of tube ID gauging and dent detection conducted in San Onofre Unit 1 steam generators demonstrate that the denting process has been arrested. Continued assurance of this condition will be provided by monitoring for dent progression as part of the general steam generator tube inspection in accordance with Specification A. Progression of denting is adequately monitored in either steam generator A or C by reviewing required eddy current probe size reductions during the performance of this inspection scope.

The results of AVB area inspections conducted in San Onofre Unit 1 steam generators demonstrate that AVB modifications installed during the Cycle VI refueling outage were successful in eliminating significant growth of tube wall penetration indications at AVB locations. Continuing assurance of this condition can be provided by performing U-bend inspections as part of the Specification A inspection scope at refueling outage intervals of tubes having wall penetration indications in excess of 20 percent at AVB locations.

TABLE 4.18.1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,c]

| Analysis | Water (pCi/l) | Airborne Particulate or Gas ($pCi/m^3$) | Marine Animals (pCi/kg, wet) | Local Crops (pCi/kg, wet) | Sediment (pCi/kg, dry) |
|---|---|---|---|---|---|
| gross beta | 4 | $1 \times 10^{-2}$ | | | |
| H-3 | 2000 | | | | |
| Mn-54 | 15 | | 130 | | |
| Fe-59 | 30 | | 260 | | |
| Co-58, 60 | 15 | | 130 | | |
| Zn-65 | 30 | | 260 | | |
| Zr-95 | 30 | | | | |
| Nb-95 | 15 | | | | |
| I-131 | 1[b] | $7 \times 10^{-2}$ | | 60 | |
| Cs-134 | 15 | $5 \times 10^{-2}$ | 130 | 60 | 150 |
| Cs-137 | 18 | $6 \times 10^{-2}$ | 150 | 80 | 180 |
| Ba-140 | 60 | | | | |
| La-140 | 15 | | | | |

c. A health physics technician# shall be on-site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent duties during this operation.

e. A Fire Brigade of at least five members shall be maintained on site at all times.# The Fire Brigade shall not include the Shift Superintendent and the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff in the following job classifications:

   1) Shift Superintendents, Control Room Supervisors, Control Operators, Assistant Control Operators, Nuclear Plant Equipment Operators, Plant Equipment Operators;

   2) Electricians and their first line supervisors;

   3) I&C Technicians, Computer Technicians, Test Technicians and their first line supervisors;

   4) Operational Health Physics Technicians and their first line supervisors;

   5) Boiler and Condenser Mechanics, Machinists, Welders, Crane Operators and their first line supervisors;

   6) Contractor or other Department personnel performing functions identical to those performed by personnel identified in items 1 through 5 above and within the organizational framework of the station.(i)

   Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel identified above work a normal 8-hour day, 40-hour week (excluding shift turnover and meal time) while the plant is operating (MODES 1, 2, 3 and 4). However, in the event that

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# The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

(i) Shift Technical Advisors are exempt from the overtime guidelines specified, since sleeping accommodations are provided.

## 6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Manager, Nuclear Training and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and 10 CFR Part 55, and shall include familiarization with relevant industry operational experience.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Manager, Station Emergency Preparedness and shall meet or exceed the requirements of National Fire Protection Association Standard No. 27-1975.