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PROCESS CONTROL PROGRAM (PCP)

1.23 The PROCESS CONTROL PROGRAM shall contain the provisions to assure that the SOLIDIFICATION of wet radioactive wastes results in a waste form with properties that meet the requirements of 10 CFR Part 61 and of low level radioactive waste disposal sites. The PCP shall identify process parameters influencing SOLIDIFICATION such as pH, oil content, H_2O content, solids content, ratio of solidification agent to waste and/or necessary additives for each type of anticipated waste, and the acceptable boundary conditions for the process parameters shall be identified for each waste type, based on laboratory scale and full-scale testing or experience. The PCP shall also include an identification of conditions that must be satisfied, based on full-scale testing, to assure that dewatering of bead resins, powdered resins, and filter sludges will result in volumes of free water, at the time of disposal, within the limits of 10 CFR Part 61 and of low level radioactive waste disposal sites.

PURGE - PURGING

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3800 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Sections 50.73 to 10 CFR Part 50. X
50.72 and

SHUTDOWN MARGIN

1.28 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. No change in part-length control element assembly position, and

b. All full-length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the area of Acceptable Operation shown on Figure 3.1-1.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside the area of Acceptable Operation shown on Figure 3.1-1, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

b. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure of 40 EFPD burnup into the current cycle.

c. At any THERMAL POWER, within 7 EFPD after reaching a core average exposure equivalent to two-thirds of the expected current cycle end-of-cycle core average burnup.

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*With Keff greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

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REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. A gravity feed flow path from either the refueling water tank or the spent fuel pool through CH-536 (RWT Gravity Feed Isolation Valve) and a charging pump to the Reactor Coolant System,

b. A gravity feed flow path from the refueling water tank through CH-327 (RWT Gravity Feed/Safety Injection System Isolation Valve) and a charging pump to the Reactor Coolant System,

c. A flow path from either the refueling water tank or the spent fuel pool through CH-164 (Boric Acid Filter Bypass Valve), utilizing gravity feed and a charging pump to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 6% delta k/k at 210°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2.1 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months when the Reactor Coolant System is at normal operating pressure by verifying that the flow path required by Specification 3.1.2.2 delivers at least 26 gpm for 1 charging pump and 63 gpm for two charging pumps to the Reactor Coolant System.

4.1.2.~~3~~2.2 The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or Mode 4 to perform the surveillance testing of Specification 4.1.2.2.b provided the testing is performed within 24 hours after achieving normal operating pressure in the reactor coolant system.

BORON DILUTION ALARMS

LIMITING CONDITION FOR OPERATION

3.1.2.7 Both startup channel high neutron flux alarms shall be OPERABLE.

APPLICABILITY: MODES 3*, 4, 5, and 6.

ACTION:

a. With one startup channel high neutron flux alarm inoperable:

   1. Determine the RCS boron concentration when entering MODE 3, 4, 5, or 6 or at the time the alarm is determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5 by either boronometer or RCS sampling.**.

b. With both startup channel high neutron flux alarms inoperable:

   1. Determine the RCS boron concentration by either boronmeter and RCS sampling** or by independent collection and analysis of two RCS samples when entering Mode 3, 4, or 5 or at the time both alarms are determined to be inoperable. From that time, the RCS boron concentration shall be determined at the applicable monitoring frequency in Tables 3.1-1 through 3.1-5, as applicable, by either boronmeter and RCS sampling** or by collection and analysis of two independent RCS samples. If redundant determination of RCS boron concentration cannot be accomplished immediately, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the method for determining and confirming RCS boron concentration is restored.

   2. When in MODE 5 with the RCS level below the centerline of the hotleg or MODE 6, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one startup channel high neutron flux alarm is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.7 Each startup channel high neutron flux alarm shall be demonstrated OPERABLE by performance of:

*Within 1 hour after the neutron flux is within the startup range following a reactor shutdown.

**With one or more reactor coolant pumps (RCP) operating the sample should be obtained from the hot leg. With no RCP operating, the sample should be obtained from the discharge line of the low pressure safety injection (LPSI) pump operating in the shutdown cooling mode.

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3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length (shutdown and regulating) CEAs, and all part-length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 6.6 inches (indicated position) of all other CEAs in its group. In addition, the position of the part length CEAs Groups shall be limited to the insertion limits shown in Figure 3.1-2A.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length CEAs inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in at least HOT STANDBY within 6 hours.

b. With more than one full-length or part-length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one or more full-length or part-length CEAs misaligned from any other CEAs in its group by more than 6.6 inches, operation in MODES 1 and 2 may continue, provided that core power is reduced in accordance with Figure 3.1-2B and that within 1 hour the misaligned CEA(s) is either:

   1. Restored to OPERABLE status within its above specified alignment requirements, or

   2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA(s) shall be aligned to within 6.6 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figures 3.1-~~3 and~~ 3.1-3 and 3.1-4; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

*See Special Test Exceptions 3.10.2 and 3.10.4.

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours.

Otherwise, be in at least HOT STANDBY within 6 hours.

d. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

e. With one part-length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 6.6 inches (indicated position) of all other part-length CEAs in its group.

f. With part length CEAs inserted beyond insertion limits, except for surveillance testing pursuant to Specification ~~4.1.3.2~~ 4.1.3.1.2, within 2 hours either: X

1. Restore the part length CEAs to within their limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by part length CEA group position using Figure 3.1-2A.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length and part-length CEA shall be determined to be within 6.6 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full-length CEA not fully inserted and each part-length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5.2 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5.2 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.*.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5.2 inches of each other at least once per 12 hours.

*CEAs are fully withdrawn (Full Out) when withdrawn to at least 144.75 inches.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System cold leg temperature less than 552°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, and (3) to ensure consistency with the FSAR safety analysis.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, ~~(3) separate flow paths, and~~ (4) an emergency power supply from OPERABLE diesel generators, The nominal capacity of each charging pump is 44 gpm at its discharge. Up to 16 gpm of this may be diverted to the volume control tank via the RCP control bleedoff. Instrument inaccuracies and pump performance uncertainties are limited to 2 gpm yielding the 26 gpm value.

With the RCS temperature above 210°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 4% delta k/k after xenon decay and cooldown to 210°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 23,800 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

With the RCS temperature below 210°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable. The restrictions of one and only one operable charging pump whenever reactor coolant level is below the bottom of the pressurizer is based on the assumptions used in the analysis of the boron dilution event.

The boron capability required below 210°F is based upon providing a 4% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 210°F to 120°F. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line.

3/4.2.2 PLANAR RADIAL PEAKING FACTORS - F_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With an F_{xy}^{m} exceeding a corresponding F_{xy}^{c}, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F_{xy}^{m}/F_{xy}^{c} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F_{xy}^{m}/F_{xy}^{c}) - 1.0]$ x 100% is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) or

c. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F_{xy}^{m}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F_{xy}^{c}), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than 40% but prior to operation above 70% of RATED THERMAL POWER, and

b. At least once per 31 Effective Full Power Days.

*See Special Test Exception 3.10.2.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within 2 hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.

b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:

1. Due to misalignment of either a part-length or full-length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.

2. Verify that the AZIMUTHAL POWER TILT is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and verify that the Variable Overpower Trip Setpoint has been reduced as appropriate within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the AZIMUTHAL POWER TILT is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

3/4.2.6 REACTOR COOLANT COLD LEG TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.2.6 The reactor coolant cold leg temperature (T_c) shall be within the Area of Acceptable Operation shown in Figure 3.2-3.

APPLICABILITY: MODE 1* and 2*#.

ACTION:

With the reactor coolant cold leg temperature exceeding its limit, restore the temperature to within its limit within 2 hours or be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.6 The reactor coolant cold leg temperature shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.4.
#With K_{eff} greater than or equal to 1

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 1815 psia and 2370 psia.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With the pressurizer pressure outside its above limits, restore the pressure to within its limit within 2 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.8 The pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

*See Special Test Exception 3.10.5

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) There are four channels, each of which is comprised of one of the four reactor trip breakers, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.~~h~~ g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full-length and part-length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to be indicated that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full-length and part-length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's inoperable and COLSS in operation, operation may continue provided that within 1 hour:

   The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, Figure 3.1-2B, otherwise Specification 3.1.3.1 remains applicable.

c. With both CEACs inoperable and COLSS out-of-service, operation may continue provided that:

1. Within 1 hour:

   a) The existing CPC value of the CPC addressable constant "BERR1" is multipled by 1.19 and the resulting value is re-entered into the CPCs.

   b) The Reactor Power Cutback System is placed out of service

   c) The COLSS out of service Limit Line, on Figure 3.2-2 of Specification 3.2.4, is not applicable to this mode of operation.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

2. Within 4 hours:

   a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 5 may be inserted no further than 127.5 inches withdrawn.

   b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to be indicated that both CEAC's are inoperable.

   c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Standby" mode except during CEA group 5 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 5 as permitted by 2.a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 6.6 inches (indicated position) of all other CEAs in its group.

4. Following a CEA misalignment with both CEAC's and COLSS inoperable, operation may continue provided that within 1 hour:

   The power is reduced to 85% of the pre-misaligned power but need not be reduced to less than 50% of RATED THERMAL POWER. This power restriction replaces the power restriction of Specification 3.1.3.1, Figure 3.1-2B otherwise Specification 3.1.3.1 remains applicable. X

ACTION 7 - With three or more auto restarts, excluding periodic auto restarts (Code 30 and Code 33), of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore an inoperable channel to OPERABLE status within 48 hours or open an affected reactor trip breaker within the next hour.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

(1) - Each STARTUP or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the linear power level, the CPC delta T power and CPC nuclear power signals to agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power sub-channel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

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

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total steady-state RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation and the ultrasonic flow meter adjusted pump curves or calorimetric calculations. X

(9) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct current values of addressable constants are installed in each OPERABLE CPC. ~~per Specification 2.2.2.~~ X

(10) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

ESFA SYSTEM FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)					
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)					
A. Sensor/Trip Units					
1. Steam Generator #2 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG1 > SG2	4	2	3	1, 2, 3	13*, 14*
B. ESFA System Logic					
1. Matrix Logic	6	1	3	1, 2, 3	17
2. Initiation Logic	4(c)	2(d)	4	1, 2, 3, 4	12
3. Manual AFAS	4(c)	2(d)	4	1, 2, 3, 4	15
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VIII. LOSS OF POWER (LOV)					
A. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	13*, 14*
B. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3	13*, 14*
IX. CONTROL ROOM ESSENTIAL FILTRATION	2	1	1	All Modes#	18*

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TABLE 3.3-3 (Continued)

TABLE NOTATIONS

(a) In MODES 3-6, the value may be decreased manually, to a minimum of 100 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(b) In MODES 3-6, the value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(c) Four channels provided, arranged in a selective two-out-of-four configuration (i.e., one-out-of-two taken twice).

(d) The proper two-out-of-four combination.

* The provisions of Specification 3.0.4 are not applicable.

~~# After the initial criticality of Unit 2 or Unit 3.~~

ACTION STATEMENTS

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6.~~h~~g. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed below.

Process Measurement Circuit	
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)
2. Steam Generator Level (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1-Low (ESF) Steam Generator Level 2-Low (ESF)

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)
2. Steam Generator Level - Low (Wide Range)	Steam Generator Level - Low (RPS) Steam Generator Level 1 - Low (ESF) Steam Generator Level 2 - Low (ESF)

STARTUP and/or POWER OPERATION may continue until the performance of the next required CHANNEL FUNCTIONAL TEST. Subsequent STARTUP and/or POWER OPERATION may continue if one channel is restored to OPERABLE status and the provisions of ACTION 14 13 are satisfied. X

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Number of Channels, operation may continue for up to 6 hours. After 6 hours operation may continue provided at least 1 train of essential filtration is in operation, otherwise, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure - Low	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ 10.6*/10.6**
2. Other CIAS actuated valves	≤ 31*/31**
3. Containment Pressure - High	
a. Safety Injection (HPSI)	≤ 30*/30**
b. Safety Injection (LPSI)	≤ 30*/30**
c. Containment Isolation	
1. CIAS actuated mini-purge valves	≤ 10.6*/10.6**
2. Other CIAS actuated valves	≤ 31*/31**
d. Main Steam Isolation	
1. MSIS actuated MSIV's	≤ 5.6*/5.6**
2. MSIS actuated MFIV's#	≤ 10.6*/10.6**
e. Containment Spray Pump	≤ 33*/23**
4. Containment Pressure - High-High	
a. Containment Spray	≤ 33*/23**
5. Steam Generator Pressure - Low	
a. Main Steam Isolation	
1. MSIS actuated MSIV's	≤ 5.6*/5.6**
2. MSIS actuated MFIV's#	≤ 10.6*/10.6**
6. Refueling Water Tank - Low	
a. Containment Sump Recirculation	≤ 45*/45**
7. Steam Generator Level - Low	
a. Auxiliary Feedwater (Motor Drive)	≤ 46*/23**
b. Auxiliary Feedwater (turbine drive)	≤ 30*/30**

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Testing of automatic actuation logic shall include energization/deenergization of each initiation relay and verification of proper operation of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays listed below are exempt from testing during POWER OPERATION but shall be tested at least once per 18 months during REFUELING and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

ACTUATION DEVICES THAT CANNOT BE TESTED AT POWER

TRAIN A ESF FUNCTION	TRAIN A ACTUATION DEVICE	TRAIN B ESF FUNCTION	TRAIN B ACTUATION DEVICE
SIAS A	K108	SIAS B	K108
SIAS A	K409	SIAS B	K409
CIAS A	K202	CIAS B	K204
CIAS A	K204	~~CIAS B~~	~~K205~~
~~CIAS A~~	~~K205~~	CSAS B	K304
CSAS A	K304	MSIS B	K305
MSIS A	K305	MSIS B	K404
MSIS A	K404	AFAS 1B	K113
AFAS 1A	K211	AFAS 1B	K211
AFAS 2A	K112	AFAS 2B	K112

In the case of the following relays which are tested during power operation, one or more pieces of equipment cannot be actuated, but can be racked out, bypassed or etc., which will not preclude the relay from being tested but will not actuate the locked out equipment associated with the relay:

SIAS A	K401	SIAS B	K301
SIAS A	K410	SIAS B	K308
SIAS A	K412	CIAS B	K203
CIAS A	K203	CIAS B	K210
CIAS A	K210	RAS B	K104
RAS A	K104	RAS B	K312
RAS A	K312	RAS B	K405
RAS A	K405		
AFAS 1A	K113		

X

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Area Monitors					
A. Fuel Pool Area RU-31	1	**	$\leq$15mR/hr	10^{-1} to 10^{4}mR/hr	22 & 24
B. New Fuel Area RU-19	1	*	$\leq$15mR/hr	10^{-1} to 10^{4}mR/hr	22.
C. Containment RU-148 & RU-149	2	1,2,3,4	$\leq$10R/hr	1R/hr to 10^{7}R/hr	27
D. Containment Power Access Purge Exhaust RU-37 & RU-38	1	#	$\leq$2.5mR/hr	10^{-1} to 10^{4}mR/hr	25
E. Main Steam					
1) RU-139 A&B	1	1,2,3,4	##	10^{-3} to 10^{4}R/hr	27
2) RU-140 A&B	1	1,2,3,4	##	10^{-3} to 10^{4}R/hr	27
2. Process Monitors					
A. Containment Building Atmosphere RU-1	2	1,2,3,4			23 & 27
1) Particulate			$\leq 2.3\times10^{-6}\mu$Ci/cc Cs-137	10^{-9} to $10^{-4}\mu$Ci/cc	
2) Gaseous			$\leq 6.6\times10^{-2}\mu$Ci/cc Xe-133	10^{-6} to $10^{-1}\mu$Ci/cc	
B. Noble Gas Monitors Control Room Ventilation Intake RU-29 & RU-30	1	ALL MODES	$\leq 2\times10^{-5}\mu$Ci/cc	10^{-6} to $10^{-1}\mu$Ci/cc	26
3. Post Accident Sampling System	1###	1,2,3	N.A.	N.A.	28

*With fuel in the storage pool or building.
**With irradiated fuel in the storage pool.
#When purge is being used.
##Three (3) times background in Rem/hour.
###The Minimum Channels Operable will be defined in the Preplanned Alternate Sampling Program. ~~of Specification 6.16.~~

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.5.1.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12 or operate the fuel building essential ventilation system while handling irradiated fuel.

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the ~~recirculation~~ essential filtration mode of operation.

ACTION 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1. For area monitors RU-139 A and B, RU-140 A and B, RU-148 and RU-149, initiate a preplanned alternate program to monitor the appropriate parameters.

2. For process monitors, place moveable air monitor in-line.

3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

ACTION 28 - With the number of OPERABLE Channels one less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 7 days, or:

1. Initiate the Preplanned Alternate Sampling Program ~~of Specification 6.16~~ to monitor the appropriate parameter(s).

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

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REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown system disconnect switches, power, controls and monitoring instrumentation channels shown in Table 3.3-9 A-C shall be OPERABLE:

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the number of OPERABLE remote A-C shutdown monitoring channels less than required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

b. With one or more remote shutdown system disconnect switches or power or control circuits inoperable, restore the inoperable switch(s)/ circuit(s) to OPERABLE status or issue procedure changes per Specification 6.8.3 that identifies alternate disconnect methods or power or control circuits for remote shutdown within 7 days, or be in HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 The Remote Shutdown System shall be demonstrated operable:

a. By performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6 for each remote shutdown monitoring instrumentation channel.

b. By operation of each remote shutdown system disconnect switch and power and control circuit including the actuated components at least once per 18 months.

CONTROLLED BY USER

TABLE 3.3-9A

REMOTE SHUTDOWN INSTRUMENTATION, ~~DISCONNECT SWITCHES AND CONTROL CIRCUITS~~

INSTRUMENTATION	READOUT LOCATION	MINIMUM CHANNELS OPERABLE
1. Log Neutron Power Level	Remote Shutdown Panel	2
2. Reactor Coolant Hot Leg Temperature	Remote Shutdown Panel	1/loop
3. Reactor Coolant Cold Leg Temperature	Remote Shutdown Panel	1/loop
4. Pressurizer Pressure	Remote Shutdown Panel	1
5. Pressurizer Level	Remote Shutdown Panel	2
6. Steam Generator Pressure	Remote Shutdown Panel	2/steam generator
7. Steam Generator Level	Remote Shutdown Panel	2/steam generator
8. Refueling Water Tank Level	Remote Shutdown Panel	2
9. Charging Line Pressure	Remote Shutdown Panel	1
10. Charging Line Flow	Remote Shutdown Panel	1
11. Shutdown Cooling Heat Exchanger Temperatures	Remote Shutdown Panel	2
12. Shutdown Cooling Flow	Remote Shutdown Panel	2
13. Auxiliary Feedwater Flow Rate	Remote Shutdown Panel	2/steam generator

CONTROLLED BY USER

TABLE 3.3-9B

REMOTE SHUTDOWN DISCONNECT SWITCHES

DISCONNECT SWITCHES	SWITCH LOCATION
1. SG 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-178A and SGB-HY-178R	RSP
2. SG 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-185A and SGB-HY-185R	RSP
3. Auxiliary Spray Valve CHB-HV-203	RSP
4. Letdown to Regenerative Heat Exchanger Isolation, CHB-UV-515	RSP
5. Reactor Coolant Pump Controlled Bleedoff, CHB-UV-505	RSP
6. Auxiliary Feedwater Pump B to SG 1 Control Valve, AFB-HV-30	RSP
7. Auxiliary Feedwater Pump B to SG 2 Control Valve, AFB-HV-31	RSP
8. Auxiliary Feedwater Pump B to SG 1 Block Valve, AFB-UV-34	RSP
9. Auxiliary Feedwater Pump B to SG 2 Block Valve, AFB-UV-35	RSP
10. Pressurizer Backup Heaters Banks B10, B18, A05 Control	RSP
11. Safety Injection Tank 2A Vent Control SIB-HV-613	RSP
12. Safety Injection Tank 2B Vent Control SIB-HV-623	RSP
13. Safety Injection Tank 1A Vent Control SIB-HV-633	RSP
14. Safety Injection Tank 1B Vent Control SIB-HV-643	RSP
15. Safety Injection Tank Vent Valves Power Supply SIB-HS-18A	RSP
16. SG 1 line 2 ~~ADV~~ Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-178B and SGD-HY-178S	RSP
17. SG 2 line 1 ~~ADV~~ Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-185B and SGD-HY-185S	RSP
18. Control BLDG Battery Room D Essential Exhaust Fan 'HJB-J01A'	PHB-M3205
19. Control BLDG Battery Room B Essential Exhaust Fan 'HJB-J01B'	PHB-M3205
20. Battery Charger D Control Room Circuits PKD-H14	PHB-M3209 AND PKD-H14
21. ESF Switchgear Room Essential AHU HJB-Z03	PHB-M3205
22. LPSI Pump SIB-P01 Breaker Control	PBB-S04F
23. Diesel Generator B Breaker Control	PBB-S04B
24. Essential Spray Pond Pump SPB-P01 Breaker Control	PBB-S04C

TABLE 3.3-9B (continued)

REMOTE SHUTDOWN DISCONNECT SWITCHES

	DISCONNECT SWITCHES	SWITCH LOCATION
25.	Essential Chiller ECB-P01 (E) Breaker Control	PBB-S04G
26.	E-PBB-S04J-4.16KV Feeder Breaker to Load Center L-32 (480V; PGB-)	PBB-S04J
27.	E-PBB-S04H-4.16KV Feeder Breaker to Load Center L-34 (480V; PGB-)	PBB-S04H
28.	E-PBB-S04N-4.16KV Feeder Breaker to Load Center L-36 (480V; PGB-)	PBB-S04N
29.	Auxiliary Feedwater Pump AFB P01 Breaker Control	PBB-S04S
30.	Essential Cooling Water Pump EWB-P01 Breaker Control	PBB-S04M
31.	E-PGB-L32B2-480V Main Feeder Breaker to Load Center L-32 (Supply; PGB-)	PGB-L32B2
32.	E-PGB-L34B2-480V Main Feeder Breaker to Load Center L-34 (Supply; PGB-)	PGB-L34B2
33.	E-PGB-L36B2-480V Main Feeder Breaker to Load Center L-36 (Supply; PGB-)	PGB-L36B2
34.	Charging Pump No. 2 CHB-P01 Supply Breaker CHB-P01	PGB-L32C1
35.	Diesel Engine Control Switch HS-2A	DGB-C01
36.	Diesel Engine Control Switch HS-2B	DGB-C01
37.	Diesel Generator Control Switch HS-2	DGB-C01
38.	Diesel Generator Essential Exhaust Fan HDB-J01	DGB-C01
39.	Diesel Generator Fuel Oil Transfer Pump DFB-P01	DGB-C01
40.	Battery Charger BD Control Room Circuits-PKB-H16	PHB-M3425
41.	Battery Charger B Control Room Circuits PKB-H12	PHB-M3627
42.	125 VDC Battery B Breaker Control Room Circuits	PKB-M4201
43.	125 VDC Battery D Breaker Control Room Circuits	PKD-M4401
44.	CS Pump B Discharge to SD HX B SIB-HV-689	PHB-M3804
45.	Shutdown Cooling LPSI Suction SIB-HV-656	PHB-M3611
46.	LPSI-CS- from SD HX B X-Tie SIB-HV-695	PHB-M3810
47.	Shutdown Cooling Warmup Bypass SIB-HV-690	PHB-M3806
48.	LPSI-CS to SD HX B Crosstie SIB-HV-694	PHB-M3416
49.	SD HX "B" to RC Loops 2A/2B SIB-HV-696	PHB-M3416

TABLE 3.3-9B(continued)

~~REMOTE SHUTDOWN DISCONNECT SWITCHES~~

DISCONNECT SWITCHES	SWITCH LOCATION
50. LPSI-SD HX "B" Bypass B SI-HV-307	PHB-M3803
51. LPSI Pump "B" Recirc B SI-UV-668	PHB-M3611
52. LPSI Pump "B" Suction from RWT SI-HV-692 B	PHB-M3805
53. SD Cooling LPSI Pump "B" Suction SI-UV-652 B	PHB-M3611
54. SD Cooling LPSI Pump "B" Suction SI-UV-654 B	PKD-B44
55. LPSI Header "B" to RC Loop 2A SI-UV-615 B	PHB-M3611
56. LPSI Header "B" to RC Loop 2B SI-UV-625 B	PHB-M3640
57. VCT Outlet Isolation CHN-HB-501 UV	NHN-M7208
58. RWT Gravity Feed CHN-HV-536	NHN-M7209
59. Shutdown Cooling Temperature Control SIB-HV-658	PHB-M3416
60. Shutdown Cooling Heat Exchanger Bypass Valve SIB-HV-693	PHB-M3416
61. 4.16 KV Bus PBB-S04 Feeder from XFMR NBN-X04	PBB-S04K
62. 4.16 KV Bus PBB-S04 Feeder from XFMR NBN-X03	PBB-S04L
63. Electrical Penetration Room B ACU HAB-Z06	PHB-M3640
64. Control Room HVAC Isolation Dampers HJB-M01/HJB-M55	RSP
65. O.S.A. Supply Damper HJB-M02	RSP
66. O.S.A. Supply Damper HJB-M03	RSP
67. R.C.S. Sample Isolation Valve SSA-UV-203	RSPA
68. R.C.S. Sample Isolation Valve SSB-UV-200	SSB-J04
69. 125 VDC Battery A Breaker Control Room Circuits	PKA-M4101

REMOTE SHUTDOWN CONTROL CIRCUITS

		CONTROL CIRCUITS	SWITCH LOCATION
	1.	Auxiliary Feedwater Pump B to S/G 1 Isolation Valve AFB-UV34 –	RSP
	2.	Auxiliary Feedwater Pump B to S/G 1 Control Valve AFB-HV30 –	RSP
	3.	Auxiliary Feedwater Pump B to S/G 2 Isolation Valve AFB-UV35 –	RSP
	4.	Auxiliary Feedwater Pump B to S/G 2 Control Valve AFB-HV31 –	RSP
	5.	Auxiliary Feedwater Pump AFB-P01	PBB-S04S
	6.	Charging Pump No. 2 CHB-P01	PGB-L32C4
	7.	Pressurizer Auxiliary Spray Valve CHB-HV203 –	RSP
	8.	Pressurizer Backup Heater Bank	RSP
	9.	Letdown to Regen HX Isolation Valve CHB-UV515 –	RSP
	10.	RCP Cont Bleedoff Valve CHB-UV505 –	RSP
	11.	Volume Control Tank Outlet Isolation Valve CHN-UV501 –	NHN-M7208
	12.	RWT Gravity Feed isolation Valve CHE-HV536 –	NHN-M7209
	13.	S/G 1 line 2 Atmospheric Dump Valve Controller SGB-HIC 178B	RSP
	14.	S/G 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-178A and SGB-HY-178R	RSP
	15.	S/G 1 line 2 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-178B and SGB-HY-178S	RSP
	16.	S/G 2 line 2 Atmospheric Dump Valve Controller SGB-HIC-185B	RSP
	17.	S/G 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-185A and SGB-HY-185R	RSP
	18.	S/G 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGB-HY-185B and SGB-HY-185S	RSP
	19.	Diesel Generator B Output Breaker	PBB-S04B
	20.	Diesel Generator Building Essential Exhaust Fan HDB-J01	DGB-B01
	21.	Diesel Generator B Fuel Oil Transfer Pump DFB-P01	DGB-B01
E-PBB-S04H	22.	4.16 KV to 480V LC-L34 Supply Breaker — Load Center PGB- Feeder Breaker	PBB-S04H
E-PBB-S04J	23.	4.16KV to 480V LC-L32 Supply Breaker — Load Center PGB- Feeder Breaker	PBB-S04J
E-PBB-S04N	24.	4.16KV to 480V LC-L36 Supply Breaker — Load Center PGB- Feeder Breaker	PBB-S04N
E-PGB-L32B2	25.	480V LC-L32 Supply Breaker — Main Supply Breaker To Load Center PGB-L32	PGB-L32B2
E-PGB-L34B2	26.	480V LC-L34 Supply Breaker — Main Supply Breaker To Load Center PGB-L34	PGB-L34B2

TABLE 3.3-9C (continued)

~~REMOTE SHUTDOWN CONTROL CIRCUITS~~

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CONTROL CIRCUITS	SWITCH LOCATION
27. 480V ~~LC-L36~~ Supply Breaker To Load Center PGB-L36	PGB-L36B2
28. Battery Charger PKB-H12 Supply Breaker	PHB-M3627
29. Battery Charger PKD-H14 Supply Breaker	PHB-M3209
30. Backup Battery Charger PKB-H16 Supply Breaker	PHB-M3425
31. Essential Spray Pond Pump SPB-P01	PBB-S04C
32. Essential Cooling Water Pump EWB-P01	PBB-S04M
33. Essential Chilled Water Chiller ECB-E01	PBB-S04G
34. Battery Room D Essential Exhaust Fan HJB-J01A	PHB-M3206
35. Battery Room B Essential Exhaust Fan HJB-J01B	PHB-M3207
36. ESF Switchgear Room B Essential AHU HJB-Z03	PHB-M3203
37. Electrical Penetration Room B ACU Fan HAB-Z06	PHB-M3631
38. SIT Vent Valves Power Supply SIB-HS-18A	RSP
39. SIT 2A Vent Valve SIB-HV613 -	RSP
40. SIT 2B Vent Valve SIB-HV623 -	RSP
41. SIT 1A Vent Valve SIB-HV633 -	RSP
42. SIT 1B Vent Valve SIB-HV643 -	RSP
43. LPSI Pump B SIB-P01	PBB-S04F
44. Containment Spray Pump B Discharger to SD HX "B" Valve SIB-HV689 -	PHB-M3804
45. LPSI Containment Spray from SD HX "B" X-tie Valve SIB-HV695 -	PHB-M3810
46. Shutdown Cooling LPSI Suction Valve SIB-HV656-	PHB-M3605
47. Shutdown Cooling Warmup Bypass Valve SIB-UV690-	PHB-M3806
48. LPSI Containment Spray to SD HX "B" X-tie Valve SIB-HV694 -	PHB-M3414

TABLE 3.3-9C (continued)

~~REMOTE SHUTDOWN CONTROL CIRCUITS~~

CONTROL CIRCUITS	SWITCH LOCATION	
49. SD HX "B" to RC Loops 2A/2B Valve SIB-HV696 -	PHB-M3415 -	X
50. LPSI SD HX "B" Bypass Valve SIB-HV307 -	PHB-M3803	X
51. LPSI Pump B Recirc. Valve SIB-UV688 -	PHB-M3609	X
52. LPSI Pump B Suction From RWT SIB-HV692 -	PHB-M3805	X
53. RC Loop to Shutdown Cooling Valve SIB-UV652 -	PHB-M3604	X
54. RC Loop to Shutdown Cooling Valve SIB-UV654 -	PKD-B44	X
55. LPSI Header B to RC Loop 2A Valve SIB-UV615 -	PHB-M3606	X
56. LPSI Header B to RC Loop 2B Valve SIB-UV625 -	PHB-M3621	X
57. SDC "B" Temperature Control Valve SIB-HV-658	PHB-M3412	
58. Control Room Ventilation Isolation Dampers HJB-M01/HJB-M55	RSP RSP	X X
59. O.S.A. Supply Damper HJB-M02	RSP	X
60. O.S.A. Supply Damper HJB-M03		
61. Diesel Generator "B" Emergency Start	DGB-B01	X
62. Normal Offsite Power Supply Breaker	PBB-S04K	
63. Alternate Offsite Power Supply Breaker	PBB-S04L	
64. Battery "B" Breaker	PKB-M4201	
65. Battery "D" Breaker	PKD-M4401	
66. RCS Sample Isolation Valve SSA-UV-203	SSA-J04	X
67. RCS Sample Isolation Valve SSB-UV-200	SSB-J04	X
68. Train "B" Pumps Combined Recirc to RWT Valve SIB-UV-659	RSP	
69. Shutdown Cooling Heat Exchanger Bypass Valve SIB-HV-693	PHB-M3413	
70. Battery "A" Breaker	PKA-M4101	

TABLE 3.3-10

ACCIDENT

POST-MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Containment Pressure	2	1	29,30
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	2	1/loop	29,30
3. Reactor Coolant Inlet Temperature - T_{cold} (Wide Range)	2	1/loop	29,30
4. Pressurizer Pressure - Wide Range	2	1	29,30
5. Pressurizer Water Level	2	1	29,30
6. Steam Generator Pressure	2/steam generator	1/steam generator	29,30
7. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator	29,30
8. Refueling Water Storage Tank Water Level	2	1	29,30
9. Auxiliary Feedwater Flow Rate	2	1	29,30
10. Reactor Cooling System Subcooling Margin Monitor	2	1	29,30
11. Pressurizer Safety Valve Position Indicator	1/valve	1/valve	29,30
12. Containment Water Level (Narrow Range)	2	1	29,30
13. Containment Water Level (Wide Range)	2	1	29,30
14. Core Exit Thermocouples	4/core quadrant	2/core quadrant	29,30
15. Reactor Vessel Water Level	2*	1*	31,32
16. Neutron Flux Monitor (Power Range)	2	1	29,30

*A channel is eight sensors in a probe. A channel is OPERABLE if four or more sensors, two or more in the upper four and two or more in the lower four, are OPERABLE.

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TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature -T_{cold} (Wide Range)	M	R
4. Pressurizer Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Wide Range	M	R
8. Refueling Water Storage Tank Water Level	M	R
9. Auxiliary Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. Pressurizer Safety Valve Position Indicator	M	R
12. Containment Water Level (Narrow Range)	M	R
13. Containment Water Level (Wide Range)	M	R
14. Core Exit Thermocouples	M	R
15. Reactor Vessel Water Level	M	R
16. Neutron Flux Monitor (Power Range)	M	R

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3.3-10
TABLE ~~4.3-7~~
ACTION STATEMENTS

ACTION 29 - With the number of OPERABLE Channels one less than the Required Number of Channels in Table 3.3-10, either restore the Inoperable Channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 30 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the Inoperable Channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 31 - With the number of OPERABLE Channels one less than the Required Number of Channels, either restore the system to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 32 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE in Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory;

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the system to OPERABLE status at the next scheduled refueling.

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER EVACUATION SYSTEM			
A. Low Range Monitors			
a. Noble Gas Activity Monitor #RU-141	1	1, 2, 3, 4 ***	37
b. Iodine Sampler	1	1, 2, 3, 4	40
c. Particulate Sampler	1	1, 2, 3, 4	40
d. Flow Rate Monitor	1	1, 2, 3, 4	36
e. Sampler Flow Rate Measuring Device	1	1, 2, 3, 4	36
B. High Range Monitors			
a. Noble Gas Activity Monitor #RU-142	1	1, 2, 3, 4 ***	42
b. Iodine Sampler	1	1, 2, 3, 4	42
c. Particulate Sampler	1	1, 2, 3, 4	42
d. Sampler Flow Rate Measuring Device	1	1, 2, 3, 4	42
4. PLANT VENT SYSTEM			
A. Low Range Monitors			
a. Noble Gas Activity Monitor #RU-143	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	1	*	36
e. Sampler Flow Rate Measuring Device	1	*	36

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TABLE 3.3-13 (Continued)

TABLE NOTATION

whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary boiler(s).

* At all times.
** During GASEOUS RADWASTE SYSTEM operation.
***# During waste gas release.
\## In MODES 1, 2, 3, and 4 or when irradiated fuel is in the fuel storage pool.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - 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.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the actions of (A) a or (B) b or (c) are performed:

a. Initiate the Preplanned Alternate Sampling Program ~~of Specification 6.16~~ to monitor the appropriate parameter(s).

b. Place moveable air monitors in-line. ~~or~~

c. take grab samples at least once per 12 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of the GASEOUS RADWASTE SYSTEM may continue provided grab samples are taken and analyzed daily. With both channels inoperable operation may continue provided grab samples are taken and analyzed (1) every 4 hours during degassing operations, and (2) daily during other operations.

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TABLE 3.3-13 (Continued)

TABLE NOTATION

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the effected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 within one hour after the channel has been declared inoperable.

ACTION 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirements, comply with the ACTION b of Specification 3.9.12 or operate the fuel building essential ventilation system while moving irradiated fuel.

ACTION 42 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement restore the channel to OPERABLE status within 72 hours or:

a. Initiate the Preplanned Alternate Sampling Program ~~of Specification 6.16~~ to monitor the appropriate parameter(s) when it is needed. X

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action(s) taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
3. CONDENSER EVACUATION SYSTEM (RU-141 and RU-142)					
a. Noble Gas Activity Monitor	D(6)	M	R(3)	Q(2)	1, 2, 3, 4 *** ***
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Flow Rate Monitor	D(7)	N.A.	R	Q	1, 2, 3, 4
e. Sampler Flow Rate Measuring Device	D(7)	N.A.	R	Q	1, 2, 3, 4
4. PLANT VENT SYSTEM (RU-143 and RU-144)					
a. Noble Gas Activity Monitor	D(6)	M	R(3)	Q(2)	*
b. Iodine Sampler	N.A.	N.A.	N.A.	N.A.	*
c. Particulate Sampler	N.A.	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D(7)	N.A.	R	Q	*
e. Sampler Flow Rate Measuring Device	D(7)	N.A.	R	Q	*

X

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

Whenever the condenser air removal system is in operation, or whenever turbine glands are being supplied with steam from sources other than the auxiliary boiler(s).

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

***

\# During waste gas release.

\## During MODES 1, 2, 3 or 4 or with irradiated fuel in the fuel storage pool.

\### Functional test should consist of, but not be limited to, a verification of system isolation capability by the insertion of a simulated alarm condition.

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway occurs if the instrument indicates measured levels above the alarm/trip setpoint.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1. Instrument indicates measured levels above the alarm setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

(6) The channel check for channels in standby status shall consist of verification that the channel is "on-line and reachable."

(7) Daily channel check not required for flow monitors in standby status.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

Response time testing of resistance temperature devices, which are a part of the reactor protective system, shall be performed by using in-situ loop current test techniques or another NRC approved method.

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications of power level, RCS flow rate, axial flux shape, radial peaking factors and CEA deviation penalties. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1 and 6.8.1) ensure that inadvertent misloading of addressable constants into the CPCs is unlikely.

Any modifications which are made to the core protection calculator software (including changes of algorithms and fuel cycle specific data) shall be performed in accordance with "CPC Protection Algorithm Software Change Procedure," CEN-39(A)-P, Revisions 2 and Supplement 1-P, Revision 01 or another NRC approved procedure on CPC software modifications.. X

CPC modifications which result in a) an unreviewed safety questions, b) a Technical Specification change, or c) methodology not previously approved by the NRC, including additions or deletions to addressable constants or modifications to the approved constant limit values, will require NRC approval prior to implementation. X

The design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs in conjunction with plant Technical Specifications will use DNBR and LPD penalty factors and increased DNBR and LPD margin to restrict reactor operation to a power level that will ensure safe operation of the plant. If the margins are not maintained, a reactor trip will occur.

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REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The value of the DNBR in Specification 2.1 is conservatively compensated for measurement uncertainties. Therefore, the actual RCS total flow rate determined by the reactor coolant pump differential pressure instrumentation or by calorimetric calculations does not have to be conservatively compensated for measurement uncertainties.

An analysis was done to specify a minimum power level below which an additional power reduction is unnecessary even if there is a CEA misalignment with CEACs out of service.

~~This work is the completion of the CEAC's Out of Service (OOS) work. This analysis improves ANPP Unit 1, Cycle 1 power capability from about 75% to greater than about 90% with both CEACs out of service.~~

The analysis determined a Power Operating Limit (POL) power and assumed a CEA misalignment occurred from this power level. The power penalty factor that would accommodate changes in radial peaks and one hour xenon redistribution that would occur if there were a CEA misalignment with CEACs out of service. The quotient of the POL power and the CEA misalignment Power Penalty factor is the maximum power (50% power) at which DNBR SAFDL violation will occur even if there is a CEA misalignment from POL conditions. Below this power, extra thermal margin will be available to the plant. Thus, for CEA misalignment, power reduction below this limiting power is unnecessary.

The lowest core power for a POL was calculated to be 70% of rated power. This was based on the following worst COLSS fluid conditions.

High Temperature :	580°F
Low Pressure :	1785 psia
ASI :	-.3
Underflow fraction:	0.865
Low Flow :	95% of full flow
High Radial Peak :	1.70 (Bank 5+4+PLR; PDIL = 40% Power)

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

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. The response times in Table 3.3-2 are made up of the time to generate the trip signal at the detector (sensor response time) and the time for the signal to interrupt power to the CEA drive mechanism (signal or trip delay time). The response times are taken from the sequence-of-events Tables in Section 15 of CESSAR.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974 as identified in the PVNGS FSAR. The seismic instrumentation for the site is located in Table 3.3-7.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972. Wind speeds less than 0.6 MPH cannot be measured by the meteorological instrumentation.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the remote shutdown system ensures that sufficient capability is available to permit safe shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

POST-ACCIDENT MONITORING
~~FIRE DETECTION~~ INSTRUMENTATION (Continued)

In the event more than four sensors in a Reactor Vessel Level channel are inoperable, repairs may only be possible during the next refueling outage. This is because the sensors are accessible only after the missile shield and reactor vessel head are removed. It is not feasible to repair a channel except during a refueling outage when the missile shield and reactor vessel head are removed to refuel the core. If both channels are inoperable, the channels shall be restored to OPERABLE status in the nearest refueling outage. If only one channel is inoperable, it is intented that this channel be restored to OPERABLE status in a refueling outage as soon as reasonably possible.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires and that fire suppression systems, that are actuated by fire detectors, will discharge extinguishing agent in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

Fire detectors that are used to actuate fire suppression systems represent a more critically important component of a plant's fire protection program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of operable fire detectors must be greater.

The loss of detection capability for fire suppression systems, actuated by fire detectors, represents a significant degradation of fire protection for any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

The fire zones listed in Table 3.3-11, Fire Detection Instruments, are discussed in Section 9B of the PVNGS FSAR.

3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exception 3.10.3.

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HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 The reactor coolant loops listed below shall be OPERABLE and at least one of these reactor coolant loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump.

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump.

APPLICABILITY: MODE 3.#

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

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

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one reactor coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 25% indicated wide range level at least once per 12 hours.

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

#See Special Test Exception 3.10.9.

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HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one reactor coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 1 and its associated steam generator and at least one associated reactor coolant pump,**

b. Reactor Coolant Loop 2 and its associated steam generator and at least one associated reactor coolant pump,**

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

APPLICABILITY: MODE 4.#

ACTION:

a. With less than the above required reactor coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 24 hours.

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

---

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

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature (saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

#See Special Test Exception 3.10.9.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

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

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

b. The secondary side water level of at least two steam generators shall be greater than 25% indicated wide range level.

APPLICABILITY: MODE 5 with reactor coolant loops filled##.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

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

##A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 255°F during cooldown, or 295°F during heatup, unless the secondary water temperature saturation temperature corresponding to steam generator pressure) of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

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

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible.

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

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

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

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 1%.*

APPLICABILITY: MODE 4.

ACTION:

a. With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

b. The provisions of Specification 3.0.4 may be suspended for up to 12 hours for entering into and during operation in MODE 4 for purposes of setting the pressurizer code safety valves under ambient (HOT) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psia ± 1%.* .

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours with the shutdown cooling system suction line relief valves aligned to provide overpressure protection for the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

AUXILIARY SPRAY

LIMITING CONDITION FOR OPERATION

3.4.3.2 Both auxiliary spray valves shall be OPERABLE.

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

ACTION:

a. With only one of the above required auxiliary spray valves OPERABLE, restore both valves 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 none of the above required auxiliary spray valves OPERABLE, restore at least one valve to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The auxiliary spray valves shall be verified to have power available to each valve every 24 hours.

4.4.3.2.2 CH-HV-524 and CH-HV-532 shall be verified locked open at least once per 31 days.

4.4.3.2.2 The auxiliary spray valves shall be cycled at least once per 18 months.
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TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Two	Two
First Inservice Inspection	All	One
Second & Subsequent Inservice Inspection	One*	One*

TABLE NOTATION

*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators, and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b.* With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow rate indicators, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

*As a one time only extension during the power ascension program, an additional 72 hours is granted to cold shutdown. During this 72 hours if the unidentified leakage exceeds 2.0 gpm, an immediate cooldown will be initiated. The RCS leakage (Surveillance Requirement 4.4.5.2.1.c) will be calculated at least once per eight hours during this 72-hour extension.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump inventory and discharge at least once per 12 hours**.

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.5.2.2 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit**:

a. At least once per 18 months,

b.* Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d.* Within 24 hours following valve actuation due to automatic or manual action or flow through the valve,

e.* Within 72 hours following a system response to an Engineered Safety Feature actuation signal.

**The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

*The provisions of Specifications 4.4.5.2.2.b, 4.4.5.2.2.d, and 4.4.5.2.2.e are not applicable for valves UV 651, UV 652, UV 653 and UV 654 due to position indication of valves in the control room.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM CHEMISTRY

~~-CHEMISTRY-~~

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	≤ 0.10 ppm	≤ 1.00 ppm
CHLORIDE	≤ 0.15 ppm	≤ 1.50 ppm
FLUORIDE	≤ 0.10 ppm	≤ 1.00 ppm

*Limit not applicable with T_{cold} less than or equal to 250°F.

REACTOR COOLANT SYSTEM

3/4.4.7 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microcuries/gram.

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

ACTION:

for more than 48 hours during one continuous time interval or exceeding the limit line shown on figure 3.4-1, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 ~~but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.~~

be in at least HOT STANDBY with Tcold less than 500°F within 6 hours.

b. With the specific activity of the primary coolant greater than 100/E ~~1.0~~ microcuries/gram ~~DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.~~

~~c. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{cold} less than 500°F within 6 hours.~~

*With T_{cold} greater than or equal to 500°F.

MODES 1,2,3,4 and 5:

With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

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

d. With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries/gram, perform the sampling and analysis requirements of item 4.(a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days with a copy to the Director, Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, and Chief, Accident Evaluation Branch, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,
2. Fuel burnup by core region,
3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,
4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and
5. The time duration when the specific activity of the primary coolant exceeded 1 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	(a) Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	(b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period. One sample is sufficient if plant has gone through a SHUTDOWN or if transient is complete in 6 hours.	1, 2, 3

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

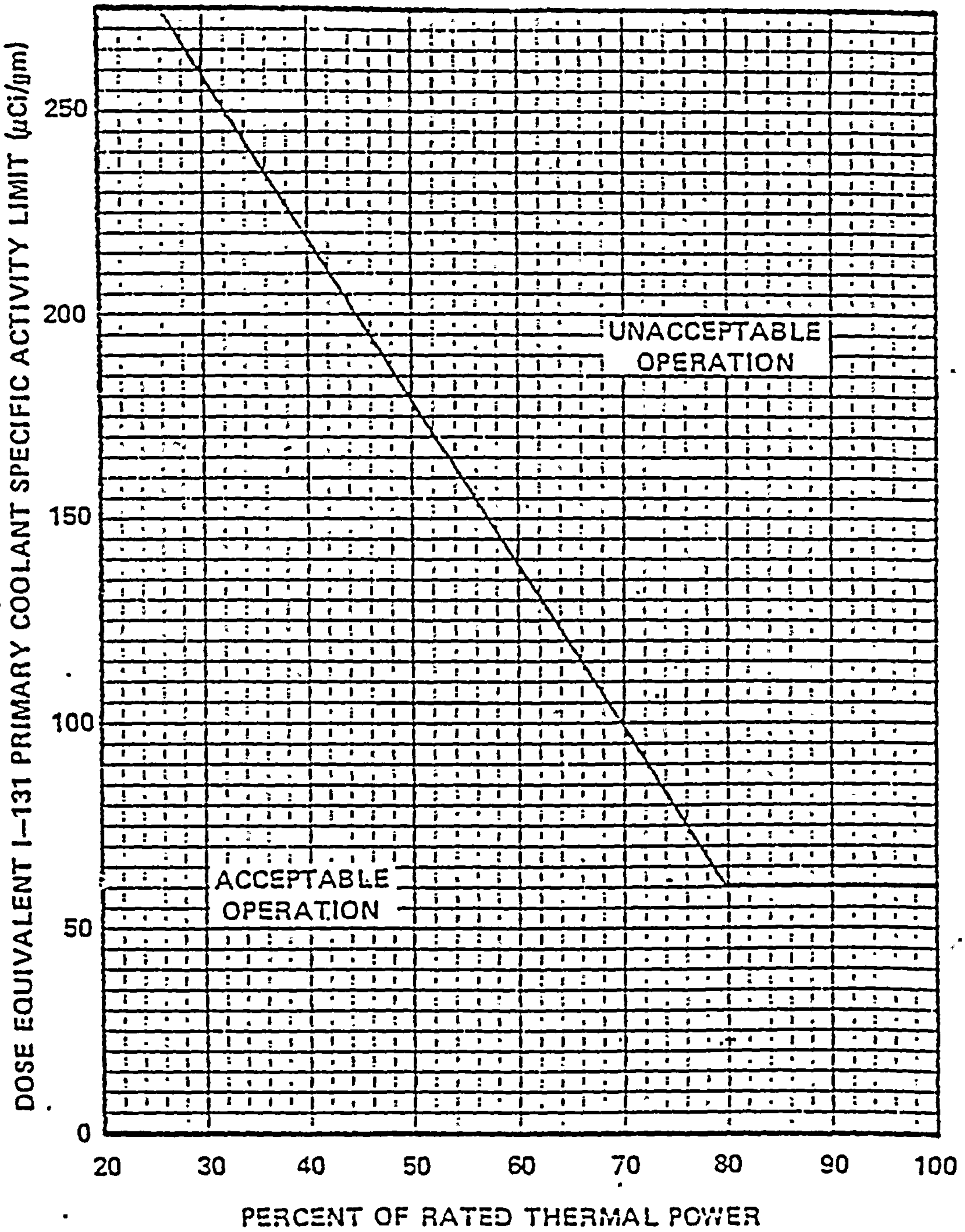

FIGURE 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY > 1.0 µCi/GRAM DOSE EQUIVALENT I-131

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate of 20°F per hour with the RCS cold leg temperature less than or equal to 95°F, 40°F per hour with RCS cold leg temperature greater than 95°F but less than or equal to 400°F, and 100°F per hour with RCS cold leg temperature greater than 400°F.

b. A maximum cooldown rate of 10°F per hour with RCS cold leg temperature less than or equal to 100°F, 40°F per hour with RCS cold leg temperature greater than 100°F but less than or equal to 130°F, and 100°F per hour with RCS cold leg temperature greater than 130°F.

c. A maximum temperature change of 10°F in any 1-hour period during inservice hydrostatic and leak testing operations.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{cold} and pressure to less than 210°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR Part 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

*See Special Test Exception 3.10.5.

FIGURE 3.4-2
RCS PRESS/TEMP LIMITS
0-10 YRS.

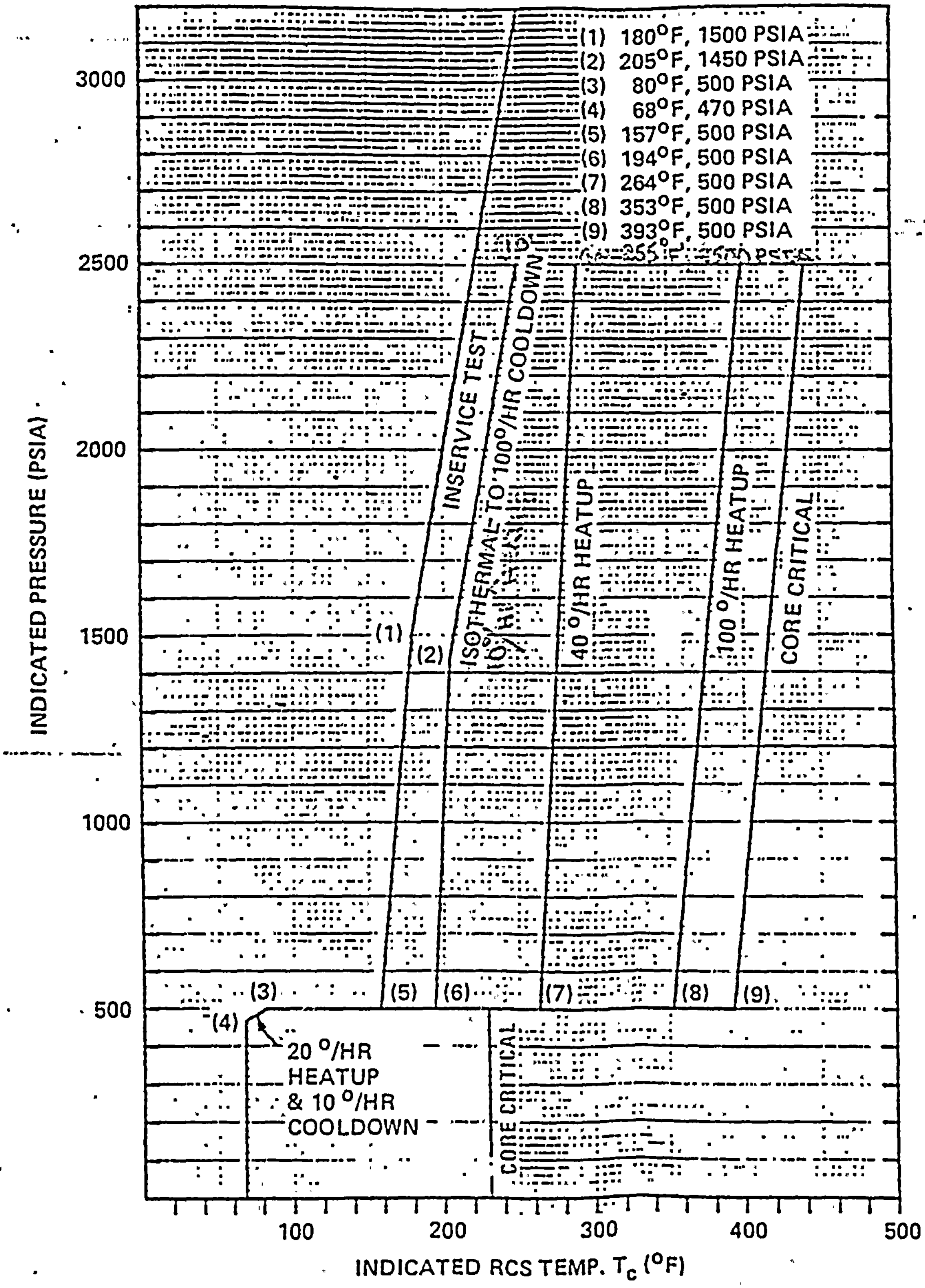

FIGURE 3/4 3.4-2

RCS PRESS/TEMP LIMITS (0 - 10 YRS) FULL POWER OPERATION

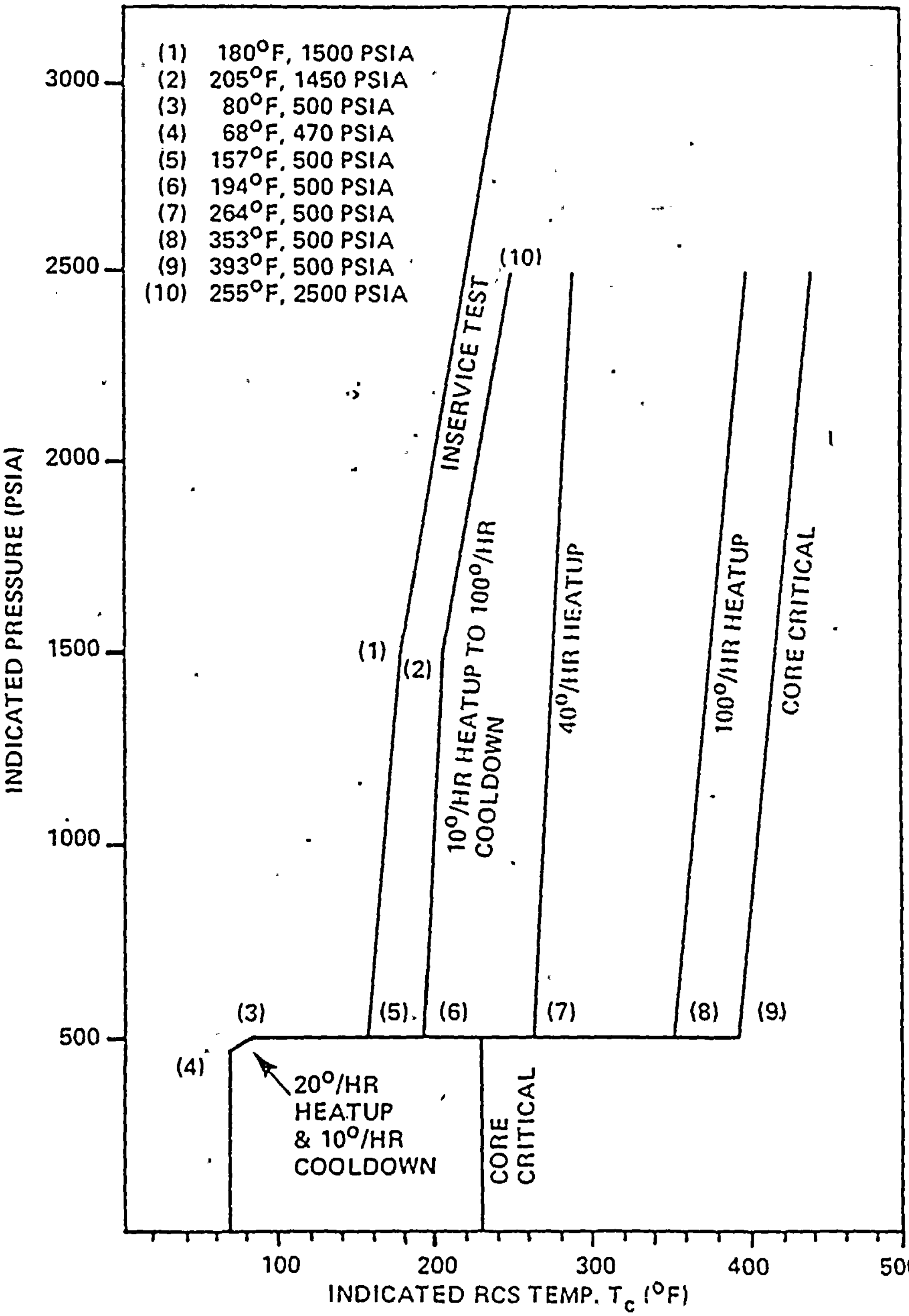

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITATIONS FOR 0 TO 10 YEARS OF FULL POWER OPERATION

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TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	38°	1.0 < LF < 1.5	8 - 10
2	43°	1.0 < LF < 1.5	Standby
3	137°	1.0 < LF < 1.5	4 - 5
4	142°	1.0 < LF < 1.5	Standby
5	230°	1.0 < LF < 1.5	12 - 15
6	310°	1.0 < LF < 1.5	18 - 24

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REACTOR COOLANT SYSTEM

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.8.2 The pressurizer temperature shall be limited to:

a. A maximum heatup rate of 200°F per hour, and

b. A maximum cooldown rate of 200°F per hour.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.8.2.1 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

4.4.8.2.2 The spray water temperature differential shall be determined for use in Table 5.7-2 for each cycle of main spray with less than four reactor coolant pumps operating and for each cycle of auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.8.3 Both shutdown cooling system (SCS) suction line relief valves with lift settings of less than or equal to 467 psig shall be OPERABLE and aligned to provide overpressure protection for the Reactor Coolant System.

APPLICABILITY: When the reactor vessel head is installed and the temperature of one or more of the RCS cold legs is less than or equal to:

a. 255°F during cooldown

b. 295°F during heatup

ACTION:

a. With one SCS relief valve inoperable, restore the inoperable valve to OPERABLE status within seven days or reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within the next eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

b. With both SCS relief valves inoperable, reduce T_{cold} to less than 200°F and, depressurize and vent the RCS through a greater than or equal to 16 square inch vent(s) within eight hours. Do not start a reactor coolant pump if the steam generator secondary water temperature is greater than 100°F above any RCS cold leg temperature.

c. In the event either the SCS suction line relief valves or an RCS vent(s) are used to mitigate an 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 initiating the transient, the effect of the SCS suction line relief valves or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 Each SCS suction line relief valve shall be verified to be aligned to provide overpressure protection for the RCS once every 8 hours during

a. Cooldown with the RCS temperature less than or equal to 255°F.

b. Heatup with the RCS temperature less than or equal to 295°F.

4.4.8.3.2 The SCS suction line relief valves shall be verified OPERABLE with the required setpoint at least once per 18 months.

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3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

shall be operable and closed at each of the following locations:

3.4.10 Both reactor coolant system vent paths ~~from the reactor vessel head shall be OPERABLE and closed.~~
a. Reactor vessel head, and
b. Pressurizer steam space.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With only one of the above required reactor coolant system ~~vessel head~~ vent paths OPERABLE, from either location restore both paths 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. at that location

b. With none of the above required reactor coolant system ~~vessel head~~ vent paths OPERABLE, from either location restore at least one path to OPERABLE status within the next 6 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours at that location

SURVEILLANCE REQUIREMENTS

4.4.10 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months, when in MODES 5 or 6, by:

a. Verifying all manual isolation valves in each vent path are locked in the open position.

b. Cycling each vent through at least one complete cycle from the control room.

c. Verifying flow through the reactor coolant system vent paths during venting.

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3/4.4.6 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.7 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss-of-offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Palo Verde site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10% of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

SPECIFIC ACTIVITY (Continued)

Reducing T_{cold} to less than 500°F prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Chapters 3 and 5 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so as not to exceed the limit lines of Figure 3.4-2. This ensures that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses at the inner wall tend to alleviate the tensile stresses induced by the internal pressure.

At the outer wall of the vessel, these thermal stresses are additive to the pressure induced tensile stresses. The magnitude of the thermal stresses at either location is dependent on the rate of heatup. Consequently, each heatup rate of interest must be analyzed on an individual basis for both the inner and outer wall.

The heatup and cooldown limit curve (Figure 3.4-2) is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curve was prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based

PRESSURE/TEMPERATURE LIMITS (Continued)

upon the fluence and residual element content, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curve Figure 3.4-2 includes predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and Appendix H of 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimens are removed and examined to determine changes in material properties. The results of these examinations shall be used to update Figure 3.4-2 based on the greater of the following:

(1) the actual shift in reference temperature ~~for plates M-6701-2 and M-4311-1 and weld 101-142~~ as determined by impact testing, or

(2) the predicted shift in reference temperature for the limiting weld and plate as determined by RG 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 40°F. The Lowest Service Temperature limit is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia. However, based upon the 10 CFR Part 50 Appendix G analysis, the isothermal condition for the reactor vessel is more restrictive than the Lowest Service Temperature line. Therefore, only the isothermal line is shown on Figure 3.4-2.

The number of reactor vessel irradiation surveillance ~~specimens~~ capsules and the frequencies for removing and testing these ~~specimens~~ capsules are provided in Table 4.4-~~3~~ 5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

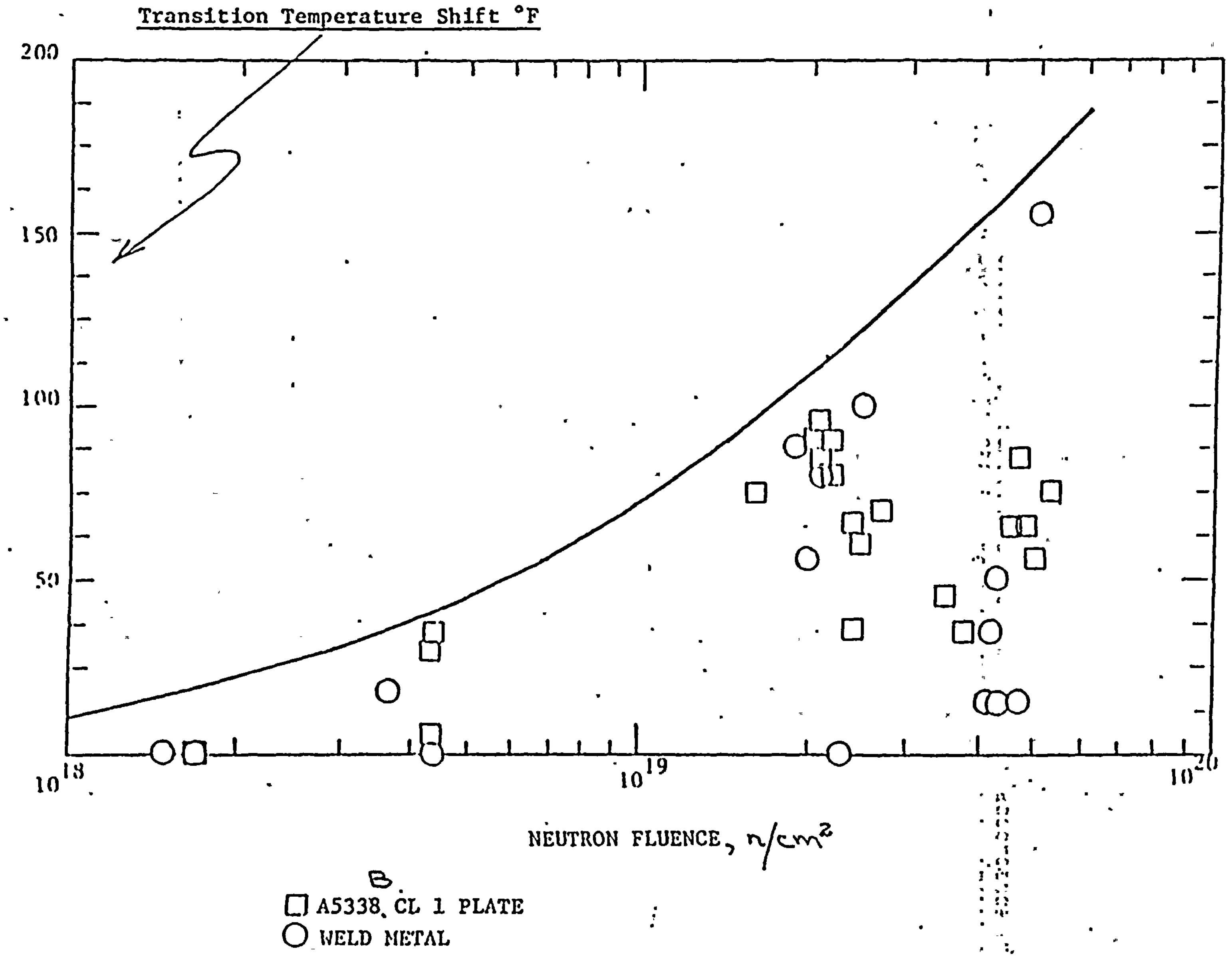

FIGURE B 3/4.4-1

NIL-DUCTILITY TRANSITION TEMPERATURE INCREASE AS A FUNCTION OF FAST (E > 1 MeV) NEUTRON FLUENCE (550°F IRRADIATION)

PRESSURE/TEMPERATURE LIMITS (Continued)

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two shutdown cooling suction line relief valves, one located in each shutdown cooling suction line, while maintaining the limits imposed on the RCS heatup and cooldown rates, ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 255°F during cooldown and 295°F during heatup. Either one of the two SCS suction line relief valves provides relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) the inadvertent safety injection actuation with two HPSI pumps injecting into a water-solid RCS with full charging capacity and with letdown isolated. These events are the most limiting energy and mass addition transients, respectively, when the RCS is at low temperatures.

The limitations imposed on the RCS heatup and cooldown rates are provided to assure low temperature overpressure protection (LTOP) with the two shutdown cooling suction line relief valves operable. ~~Figure B 3/4.4-2 provides the limits of Appendix G to 10 CFR Part 50 for various heatup and cooldown rates.~~ At low temperatures with the relief valves aligned to the RCS, it is necessary to restrict heatup and cooldown rates to assure that the P/T limits are not exceeded. During worst case transients, RCS peak pressures can reach the relief valve setpoint, 467 psig, plus accumulation. At temperatures greater than 255°F during cooldown and 295°F during heatup, the heatup and cooldown rate limitations assure the limits of Appendix G to 10 CFR 50 will not be exceeded with overpressure protection provided by the primary safety valves.

3/4.4.9 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a (g) (6) (i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve key-locked open and power to the valve removed,

b. A contained borated water level of between 1802 cubic feet (28% narrow range indication) and 1914 cubic feet (72 % narrow range indication),

c. A boron concentration between 2000 and 4400 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and 625 psig.

e. Nitrogen vent valves closed and power removed.**

f. Nitrogen vent valves capable of being operated upon restoration of power.

APPLICABILITY: MODES 1*, 2*, 3,*†, and 4*†.

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, and

†With pressurizer pressure greater than or equal to 1837 psia. When pressurizer pressure is less than 1837 psia, at least three safety injection tanks must be OPERABLE, each with a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 1415 cubic feet (60% wide range indication) and 1914 cubic feet (83% wide range indication). With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 254 psig and a maximum pressure of 625 psig, and a contained borated water volume of between 962 cubic feet (39% wide range indication) and 1914 cubic feet (83% wide range indication). In MODE 4 with pressurizer pressure less than 430 psia, the safety injection tanks may be isolated.

*See Special Test Exceptions 3.10.6 and 3.10.8.

**Nitrogen vent valves may be cycled as necessary to maintain the required nitrogen cover pressure per Specification 3.5.1d.

X

SURVEILLANCE REQUIREMENTS (Continued)

1. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

2. Verifying that a minimum total of 464 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

3. Verifying that when a representative sample of 0.055 ± 0.001 lb of TSP from a TSP storage basket is submerged, without agitation, in 1.0 ± 0.05 gallons of 77 ± 9 °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on (SIAS and RAS) test signal(s).

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection actuation test signal:

   a. High pressure safety injection pump.

   b. Low pressure safety injection pump.

3. Verifying that on a recirculation actuation test signal, the containment sump isolation valves open, the HPSI, LPSI and CS pump minimum bypass recirculation flow line isolation valves and combined SI mini-flow valve close, and the LPSI pumps stop.

4. .....

f. By verifying that each of the following pumps develops the indicated differential pressure at or greater than their respective minimum allowable recirculation flow when tested pursuant to Specification 4.0.5:

1. High pressure safety injection pump greater than or equal to 1761 psid.

2. Low pressure safety injection pump greater than or equal to 165 psid.

4. Conducting an inspection of all ECCS piping outside of containment, which is in contact with recirculation sump inventory during LOCA conditions, and verifying that the total measured leakage from piping and components is less than 1 gpm when pressurized to at least 40 psig.

X

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

LPSI System Valve Number	Hot Leg Injection Valve Number
1. SIB-UV 615, SIA-UV 306	~~1. SIA-HV-604~~
2. SIB-UV 625, SIB-UV 307	~~2. SIB-HV-609~~
3. SIA-UV 635	1. 3. SIC-HV 321
4. SIA-UV 645	2. 4. SID-HV 331

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying the following flow rates:

HPSI System - Single Pump

The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 816 gpm.

LPSI System - Single Pump

1. Injection Loop 1, total flow equal to 4900 $\pm$ 100 gpm
2. Injection Legs 1A and 1B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.
3. Injection Loop 2, total flow equal to 4900 $\pm$ 100 gpm
4. Injection Legs 2A and 2B when tested individually, with the other leg isolated, shall be within 100 gpm of each other.

Simultaneous Hot Leg and Cold Leg Injection - Single Pump

1. Hot Leg, flow equal to 545 $\pm$ 20 gpm
2. Cold Leg, flow equal to 545 $\pm$ 20 gpm

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ECCS SUBSYSTEMS (Continued)

assurance that proper ECCS flows will be maintained in the event of a LOCA.* Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

The term "minimum bypass recirculation flow," as used in Specification 4.5.2e.3. and 4.5.2f., refers to that flow directed back to the RWT from the ECCS pumps for pump protection. Testing of the ECCS pumps under the condition of minimum bypass recirculation flow in Specification 4.5.2f. verifies that the performance of the ECCS pumps supports the safety analysis minimum RCS pressure assumption at zero delivery to the RCS.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the refueling water tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water plus 10% margin is available to permit 20 minutes of engineered safety features pump operation, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

---

*The following test conditions, which apply during flow balance tests, ensure that the ECCS subsystems are adequately tested.

1. The pressurizer pressure is at atmospheric pressure.
2. The miniflow bypass recirculation lines are aligned for injection.
3. For LPSI system, (add/subtract) 6.4 gpm (to/from) the 4900 gpm requirement for every foot by which the difference of RWT water level above the RWT RAS setpoint level (exceeds/is less than) the difference of RCS water level above the cold leg centerline.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either 0.75 L_a or 0.75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either 0.75 L_a or 0.75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either 0.75 L_a or 0.75 L_t at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the Type A test by verifying that the supplemental test result L_c minus the sum of the Type A test result, L_{am}, and the superimposed leak rate, L_o, is equal to or less than 0.25 L_a.

2. Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a, 49.2 psig, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Specifications 4.6.1.7.3 and 4.6.1.7.4. X
.2 .3

f. Air locks shall be tested and demonstrated OPERABLE per Specification 4.6.1.3.

g. The provisions of Specification 4.0.2 are not applicable.

SURVEILLANCE REQUIREMENTS (Continued)

b. By conducting overall air lock leakage tests at not less than P_a, 49.2 psig, and verifying the overall air lock leakage rate is within its limit:

   1. At least once per 6 months#, and

   2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR Part 50.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.

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

ACTION:

With the containment average air temperature greater than 120°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any five of the following locations and shall be determined at least once per 24 hours:

Location

a. Elevation 85'0"

b. Elevation 85'0"

c. Elevation 126'0"

Nominal

d. Elevation 126'0"

e. Elevation 145'0"

f. Elevation 188'0"

g. Elevation 188'0"

X

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a containment spray actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

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

ACTION:

With one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable spray system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is positioned to take suction from the RWT on a containment spray actuation (CSAS) test signal.

b. By verifying that each pump develops an indicated differential pressure of greater than or equal to 273 psid at greater than or equal the minimum allowable recirculation flowrate when tested pursuant to Specification 4.0.5.

c. At least once per 31 days by verifying that the system piping is full of water to the 60 inch level in the containment spray header ($\geq$115 foot level).

d. At least once per 18 months, during shutdown, by:
   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) and recirculation actuation (RAS) test signal.
   2. Verifying that upon a recirculation actuation test signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

*Only when shutdown cooling is not in operation.

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. An spray chemical addition tank containing a level of between 90% and 100% (816 and 896 gallons) of between 33% and 35% by weight N_2H_4 solution, and

b. Two spray chemical addition pumps each capable of adding N_2H_4 solution from the spray chemical addition tank to a containment spray system pump flow.

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

ACTION:

With the iodine removal system inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the iodine removal system to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The iodine removal system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the N_2H_4 solution by chemical analysis.

c. By verifying that on recirculation flow, each spray chemical addition pump develops a discharge pressure of 100 psig when tested pursuant to Specification 4.0.5.

d. At least once per 18 months, during shutdown, by

1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation (CSAS) test signal, and

2. Verifying that each spray chemical addition pump starts automatically on a CSAS test signal.

* When the containment spray system is required to be OPERABLE.

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3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

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

ACTION:

1. With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

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

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

   c. Isolate the affected penetration within 4 hours by use of at least one closed manual valve or blind flange;*;or

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 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.

4.6.3.2 Each isolation valve specified in Sections A, B, and C of Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

   a. Verifying that on a CIAS, CSAS or SIAS test signal, each isolation valve actuates to its isolation position.

   b. Verifying that on a CPIAS test signal, all containment purge valves actuate to their isolation position.

*The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

TABLE 3.6-1 (Continued)
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)	
		A. CONTAINMENT ISOLATION (CIAS) (Continued)		
HPA-UV 001	35	Containment to hydrogen recombiner	12	
HPA-UV 003	35	Containment to hydrogen recombiner	12	
HPA-UV 024	35	H_2 control system	5	
HPB-UV 002	36	Containment to hydrogen recombiner	12	
HPA-UV 005	38	Containment to hydrogen recombiner	12	
HPB-UV 004	36	H_2 recombiner return to containment (inlet)	12	
HPA-UV 023	38	H_2 control system	5	
HPB-UV 006	39	H_2 recombiner return to containment (inlet)	12	
CHA-UV 516	40	Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger	5	
CHB-UV 523	40	Letdown line from RC loop 2B to regenerative heat exchanger and letdown heat exchanger	5	
CHB-UV 924	40	Letdown line to post-accident sampling system	5	
SSB-UV 201	42A	liquid sample Pressurizer ~~sample-surge~~ line	5	X
SSA-UV 204	42A	liquid sample Pressurizer ~~sample-surge~~ line	5	X
SSB-UV 202	42B	steam space sample Pressurizer ~~sample-surge~~ line	5	X
SSA-UV 205	42B	steam space sample Pressurizer ~~sample-surge~~ line	5	X
SSB-UV 200	42C	Hot leg sample ~~Pressurizer-sample-surge~~ line	5	X
SSA-UV 203	42C	Hot leg sample ~~Pressurizer-sample-surge~~ line	5	X

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		F. NORMALLY OPEN - ESF ACTUATED CLOSED	
SG-UV 170#	1	Main steam isolation	N.A.*
SG-UV 171#	2	Main steam isolation	N.A.*
SGE-UV 169#	1 & 2	Main steam isolation bypass	N.A.
SG-UV 180#	3	Main steam isolation	N.A.*
SG-UV 181#	4	Main steam isolation	N.A.*
SGE-UV 183#	3 & 4	Main steam isolation bypass	N.A.
SGA-UV 1133#	1-4	Steam trap/bypass	N.A.
SGA-UV 1134#	1-4	Steam trap/bypass	N.A.
SGB-UV 1135A#	1-4	Steam trap/bypass	N.A.
SGB-UV 1135B#	1-4	Steam trap/bypass	N.A.
SGB-UV 1136A#	1-4	Steam trap/bypass	N.A.
SGB-UV 1136B#	1-4	Steam trap/bypass	N.A.
SGA-UV 174#	8	Steam generator feedwater	N.A.
SGB-UV 132#	8	Steam generator feedwater	N.A.
SGB-UV 137#	10	Steam generator feedwater	N.A.
SGA-UV 177#	10	Steam generator feedwater	N.A.
SGB-UV 130#	11	Downcomer FIV	N.A.
SGA-UV 172#	11	Downcomer FIV	N.A.
SGB-UV 135#	12	Downcomer FIV	N.A.

#Not Type C tested

*Valves also covered by Specification 3/4.7.1.~~4~~ 5

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		H. NORMALLY CLOSED/POST ACCIDENT CLOSED VALVES	
SGE-V-603#	1	N_2 blanket supply/N_2 vent	N.A.
SGE-V-611#	3	N_2 blanket supply/N_2 vent	N.A.
DWE-V 061*	6	Containment demineralized water stations	N.A.
DWE-V 062*	6	Containment demineralized water stations	N.A.
FPE-V 089*	7	Fire protection containment	N.A.
SIE-V 463*	28	Safety injection drain tank	N.A.
CHE-V 854*	41	Chemical addition unit to regenerative heat exchanger	N.A.
PCE-V 070	50	Fuel pool cooling	N.A.
PCE-V 071	50	Fuel pool cooling	N.A.
PCE-V 075	51	Refueling pool cleanup	N.A.
PCE-V 076	51	Refueling pool cleanup	N.A.
IAE-V 072*	59	Containment service air utility station	N.A.

*May be opened on an intermittent basis under administrative control.

Not type C tested.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

~~c. With one hydrogen monitor inoperable, and the hydrogen monitor in the Post Accident Sampling System OPERABLE, the provisions of Specification 3.0.4 are not applicable.~~ X

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing a nominal:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

HYDROGEN PURGE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A containment hydrogen purge cleanup system, shared among the three units, shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1* and 2.*

ACTION:

With the containment hydrogen purge cleanup system inoperable and one hydrogen recombiner OPERABLE as determined by Specification 4.6.4.2, restore the hydrogen purge cleanup system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

    1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 50 scfm ± 10%.

    2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978,** meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.**

*With less than two hydrogen recombiners OPERABLE.
**ANSI N509-1980 is applicable for this specification.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 4 psig and (2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 49.2 psig. The limit of 2.5 psig for initial positive containment pressure will limit the total pressure to 49.2 psig which is less than the design pressure (60 psig) and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 49.2 psig in the event of a LOCA. The containment design pressure is 60 psig. The measurement of containment tendon lift-off force; the tensile tests of the tendon wires or strands; the examination and testing of the sheathing filler grease; and the visual examination of tendon anchorage assembly hardware, surrounding concrete and the exterior surfaces of the containment are sufficient to demonstrate this capability. The tendon wire or strand samples will also be subjected to tests. All of the required testing and visual examinations should be performed in a time frame that permits a comparison of the results for the same operating history.

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 3~~, 1984, and Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," 1984.~~ 1, 1974

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, ~~the~~ results of the engineering evaluation, and the corrective actions taken.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment automatic 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 and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The only valves in the Table 6.2.4-1 of the PVNGS FSAR that are not required to be listed in Table 3.6-1 are the following: main steam safety valves and main steam atmospheric dump valves. The main steam safety valves and the atmospheric dump valves have very high pressure setpoints to actuate and are covered by Specifications 3/4.7.1.1 and 3/4.7.1.6, respectively.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and 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 (or the purge system) is capable of controlling the expected hydrogen generation associated with (1) zirconium-water reactions, (2) radiolytic decomposition of water and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

There is a hydrogen monitor and a oxygen monitor in the Post Accident Sampling System (PASS). There is a line from the two containment hydrogen monitors to the PASS which allows the licensee to monitor hydrogen concentration in the containment following an accident with the PASS. An OPERABLE hydrogen monitor in the PASS allows the licensee to enter Modes 1 and 2 with one containment hydrogen monitor inoperable. For the hydrogen monitor in PASS to be OPERABLE, the valves in the piping to the monitor must be OPERABLE.

The use of ANSI Standard N509 (1980) in lieu of ANSI Standard N509 (1976) to meet the guidance of Regulatory Guide 1.52, Revision 2, Positions C.6.a and C.6.b, has been found acceptable as documented in Revision 2 to Section 6.5.1 of the Standard Review Plan (NUREG-0800).

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

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

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more** main steam safety valves inoperable per steam generator, operation in MODES 1, and 2, ~~and 3~~ may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the ~~Power Level-High trip setpoint is~~ Maximum Variable Overpower trip setpoint, and the Maximum Allowable Steady State Power Level are reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. Operation in MODES 3 and 4* may proceed with at least one reactor coolant loop and associated steam generator in operation, provided that there are no more than four inoperable main steam safety valves associated with the operating steam generator; otherwise, be in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*Until the steam generators are no longer required for heat removal.

**The maximum number of inoperable safety valves on any operating steam generator is four (4).

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a level of at least 23 feet (300,000 gallons).

APPLICABILITY: MODES 1, 2, 3,# and 4.*#.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or.

essential

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN with a OPERABLE shutdown cooling loop in operation within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the level (contained water volume) is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 The reactor makeup water tank shall be demonstrated OPERABLE at least once per 12 hours whenever the reactor makeup water tank is the supply source for the auxiliary feedwater pumps by verifying:

essential

a. That the reactor makeup water tank supply line to the auxiliary feedwater system isolation valve is open, and

b. That the reactor makeup water tank contains a water level of at least 26 feet (300,000 gallons).

*Until the steam generators are no longer required for heat removed.

#Not applicable when cooldown is in progress.

ATMOSPHERE DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 The atmospheric dump valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*#.

ACTION:

With less than one atmospheric dump valve per steam generator OPERABLE, restore the required atmospheric dump valve to OPERABLE status within 72 hours; or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each atmospheric dump valve shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying that the nitrogen accumulator tank is at a pressure ≥ 400 PSIG.

b. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, verify that all valves will open and close fully.

*When steam generators are being used for decay heat removal.

#See Special Test Exception 3.10.9.

3/4.7.7 CONTROL ROOM ESSENTIAL FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent control room essential filtration systems shall be OPERABLE.

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3, and 4:

With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room essential filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room essential filtration system.

b. With both control room essential filtration systems inoperable, or with the OPERABLE control room essential filtration system, required to be OPERABLE ~~in-the-recirculation-mode~~ by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room essential filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

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3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All hydraulic and mechanical snubbers shall be OPERABLE. The only snubbers excluded from this requirement are those installed on nonsafety-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.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.9 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Snubber Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all hydraulic and mechanical snubbers. If all snubbers of each type are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection of that type shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given type shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type on that system that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specifications 4.7.9f. When a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as-found setting, extending the piston rod in the tension mode direction. ~~All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.~~ Snubbers which appear inoperable during an area post maintenance inspection, area walkdown, or Transient Event Inspection shall not be considered inoperable for the purpose of establishing the Subsequent Visual Inspection Period provided that the cause of the inoperability is clearly established and remedied for that particular snubber and for the other snubbers, irrespective of type, that may be generally susceptible. X

d. Transient Event Inspection

An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems shall be made within 6 months following such an event. In addition to satisfying X

*The inspection interval for each type of snubber on a given system shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found on that system.

#The provisions of Specification 4.0.2 are not applicable.

A.C. SOURCES
CATHODIC PROTECTION
LIMITING CONDITIONS FOR OPERATION

3.8.1.3 The Cathodic Protection System associated with the Diesel Generator Fuel Oil Storage Tanks shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With Cathodic Protection System inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of malfunction and the plans for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.1.3 Verify that the Cathodic Protection System is OPERABLE at the following time intervals:

1. Verify at least once per 92 61 days that the Cathodic Protection rectifiers are OPERABLE and have been inspected in accordance with Regulatory Guide 1.137.

2. Verify at least once per 18 12 months that the Cathodic Protection is OPERABLE and providing adequate protection against corrosion in accordance with Regulatory Guide 1.137.

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3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses within the unit.

a. Train "A" A.C. emergency busses consisting of:

1. 4160-volt ESF Bus #E-PBA-S03
2. 480-volt ESF Load Center #E-PGA-L31
   a. MCC E-PHA-M31
3. 480-volt ESF Load Center #EPGA-L33
   a. MCC E-PHA-M33
   b. MCC E-PHA-M37
4. 480-volt ESF Load Center #E-PGA-L35
   a. MCC E-PHA-M35

b. Train "B" A.C. emergency busses consisting of:

1. 4160-volt ESF Bus #E-PBB-S04
2. 480-volt ESF Load Center #E-PGB-L32
   a. MCC E-PHB-M32
   b. MCC E-PHB-M38
3. 480-volt ESF Load Center #E-PGB-L34
   a. MCC E-PHB-M34
4. 480-volt ESF Load Center #E-PGB-L36
   a. MCC E-PHB-M36

c. 120-volt Channel A Vital A.C. Bus #E-PNA-D25 energized from its associated inverter connected to D.C. Channel A.*.

d. 120-volt Channel B Vital A.C. Bus #E-PNB-D26 energized from its associated inverter connected to D.C. Channel B.*.

e. 120-volt Channel C Vital A.C. Bus #E-PNC-D27 energized from its associated inverter connected to D.C. Channel C.*.

f. 120-volt Channel D Vital A.C. Bus #E-PND-D28 energized from its associated inverter connected to D.C. Channel D.*.

g. 125-volt D.C. Channel A energized from Battery Bank E-PKA-F11.

h. 125-volt D.C. Channel B energized from Battery Bank E-PKB-F12.

i. 125-volt D.C. Channel C energized from Battery Bank E-PKC-F13.

j. 125-volt D.C. Channel D energized from Battery Bank E-PKD-F14.

*Two inverters may be disconnected from their D.C. bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-M2805	E-NHN-M2827A	SG1 COLD LEG BLOWDOWN ISO VLV J-SGE-HV-41
E-NHN-M2806	E-NHN-M2827B	SG HOT LEG BLOWDOWN ISOLATION VALVE J-SGE-HV-43
E-NHN-M2827	E-NHN-M2827A	REACTOR COOL ANT PUMP OIL LIFT PUMP 1B M-RCN-P02BP
E-NHN-M2828	E-NHN-M2827A	REACTOR COOLANT PUMP OIL LIFT PUMP 2B M-RCN-P02DP
E-NHN-M2809	E-NHN-M2827C	CONTAINMENT EQUIP HATCH J-ZCN-E02
E-NHN-M2811	E-NHN-M2832A	30A RECEPTACLES FOR CTMT BLDG JIB CRANE M-ZCN-G04 A, B
E-NHN-M2818	E-NHN-M2832A	30A RECEPTACLES FOR SEAL CRANE ASSY MOT
E-NHN-M2817	E-NHN-M2832B	CTMT BLDG MONORAIL HOIST 1 TON M-ZCN-G03
E-NHN-M2819	E-NHN-M2832B	30A RECEPTACLES FOR CTMT BLDG JIB CRANE M-ZCN-G04 A, B
E-NHN-M2820	E-NHN-M2832D	CTMT BLDG ELEV #2 CONTROLLER J-ZCN-E01
E-NHN-M2821	E-NHN-M2828C	MULTIPLE STUD TENSIONER M-ZCN-M15
E-NHN-M2822	E-NHN-M2828B	WELDING RECPTS E-NHN-I09 B, C, D
E-NHN-M2801A	E-NHN-M2827B	FUEL TRANSFER SYS CONTROL CONSOLE E-PCE-D02
E-NHN-M2833	E-NHN-M2827B	REFUELING MACHINE E-PCE-J02
E-NHN-M2833A	E-NHN-M2827B	CEA CHANGE PLATFORM E-PCE-J01

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PGB-L34D2	E-NGN-B34D2 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A01D
E-PGB-L34D3	E-NGN-B34D3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02D
E-PGB-L36D3	E-NGN-B3603 (FUSE)	CTMT NOR ACU FAN M-HCN-A01B
E-PHA-M3318	E-PHA-M3334	SAFETY INJECT TANK 4 ISOL VLV J-SIA-UV-644
E-PHA-M3316	E-PHA-M3316A	SAFETY INJECT TANK 3 ISOL VLV J-SIA-UV-634
E-PHB-M3404	E-PHB-M3405B	NCWS RET INT CTMT ISOL VLV J-NCB-UV-403
~~E-PHA-M3519~~	~~E-PHA-M3521A~~	~~CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-4B~~
E-PHA-M3517	E-PHA-M3521	CTMT PRG RFL MODE ISO VLV J-CPA-UV-2B
E-PHA-M3503	E-PHA-M3507A	SHUT DN CLG ISOL LOOP 1 VLV J-SIA-UV-651
E-PHA-M3508	E-PHA-M3511A	CTMT/RAD SUMP CTMT INT ISO VLV J-RDA-UV-23
E-PHA-M3512	E-PHA-M3513A	CTMT SUMP ISOL TRAIN A VLV J-SIA-UV-673
E-PHB-M3622	E-PHB-M3629	CTMT PRG REFULING MODE ISO VLV J-CPB-UV-3A
E-PHB-M3604	E-PHB-M3604A	SHUT DN CLG ISOL LOOP 2 VLV J-SIB-UV-652
E-PHB-M3619	E-PHB-M3641A	SAFETY INJECTION TANK ISOL VLV J-SIB-UV-614
~~E-PHB-M3624~~	~~E-PHA-M3607A~~	~~CTMT PRG PWR ACCESS MODE ISO VLV J-CPB-UV-5A~~

X

X

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TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-PHB-M3613	E-PHB-M3613A	CTMT SUMP ISOL TRAIN B VLV J-SIB-UV-675
E-PHB-M3618	E-PHB-M3641	SAFETY INJECTION TANK 2 ISO VLV J-SIB-UV-624
E-PHA-M3704	E-PHA-M3703A	WASTE GAS HEADER CONTAINMENT ISOLATION VALVE GRA UV1 J-
E-PHA-M3715	E-PHA-M3719	H_2 CONT TRAIN A UPSTM SUP ISO VLV J-HPA-UV-1
E-PHB-M3816	E-PHB-M3836	H_2 CTMT TRAIN B UPSTM SUP ISO VLV J-HPB-UV-2
E-PHB-M3811	E-PHB-M3813A	NORM CHIL WTR RETURN CTMT ISO VLV J-WCB-UV-61
E-PKD-B44	E-PKD-M4411	SHUTDOWN CLG ISOL VLV -J-SID-UV-654
E-PKC-B43	E-PKC-M4311	SHUTDOWN COOLING ISOL VLV J-SIC-UV-653
E-NNN-D1113	E-NNN-D11	MOVABLE INCORE DRIVE SYS #I 800VA, M-RIN-M03A VIA E-RIN-J01A
E-NNN-D1213	E-NNN-D12	MOVABLE INCORE DRIVE SYS #II 800VA, M-RIN-M03B VIA E-RIN-J01A
E-NNN-D1526	E-NNN-D15	RCP INSTM LOCAL PNL J-RCN-E02
E-NNN-D1525	E-NNN-D15	RCP INSTM LOCAL PNL J-RCN-E01
E-NNN-D1626	E-NNN-D16	RCP INSTM LOCAL PNL J-RCN-E04
E-NNN-D1625	E-NNN-D16	RCP INSTM LOCAL PNL J-RCN-E03
E-QAN-D05B	E-QAN-B02	LIGHTING PANEL E-QAN-D05B CTMT BLDG EL 100'

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-QAN-D05C	E-QAN-B03	LIGHTING PANEL E-QAN-D05C CTMT BLDG EL 100'
E-QAN-D05D	E-QAN-B04	LIGHTING PANEL E-QAN-D05D CTMT BLDG EL 140'
E-QAN-D05F	E-QAN-B05	LIGHTING PANEL E-QAN-D05F CTMT BLDG EL 140'
E-QAN-D05E	E-QAN-B06	LIGHTING PANEL E-QAN-D05E CTMT BLDG EL 140'
E-QBN-B01	E-QBN-D91	LIGHTING PANEL E-QBN-D73A CTMT BLDG EL 100'
E-QBN-B02	E-QBN-D91	LIGHTING PANEL E-QBN-D73B CTMT BLDG EL 140'
E-NHN-D1514	E-NHN-M1526	TO OPERATION CAMERA JB# 2
E-RCN-D0101, 2	E-NGN-L11C2	PZR BU HTR M-RCE-B07, B13, A01
E-NHN-D2614	E-NHN-M2618	TO OPERATION CAMERA JB# 1
E-RCN-D0102, 1	E-NGN-L11C2	PZR BU HTR M-RCE-B03, A09, A15
E-RCN-D0302, 1	E-NGN-L11C3	PZR BU HTR M-RCE-B04, A11, A16
E-RCN-D0301, 2	E-NGN-L11C3	PZR BU HTR M-RCE-A02, A07, A13
E-RCN-D0202, 1	E-NGN-L12C2	PZR BU HTR M-RCE-B06, B12, A18
E-RCN-D0201, 2	E-NGN-L12C2	PZR BU HTR M-RCE-B16, A04, A08
E-RCN-D0402, 1	E-NGN-L12C3	PZR BU HTR M-RCE-B15, A03, A10
E-RCN-D0401, 2	E-NGN-L12C3	PZR BU HTR M-RCE-A17, A06, A12

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TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NAN-S01M	E-NAN-S01A E-NAN-S03B	RCP M-RCE-P01A (C.E. NO. 1A)
E-NAN-S01L	E-NAN-S01A E-NAN-S03B	RCP M-RCE-P01C (C.E. NO. 2A)
E-NAN-S02L	E-NAN-S02A E-NAN-S04B	RCP M-RCE-P01B (C.E. NO. 1B)
E-NAN-S02M	E-NAN-S02A E-NAN-S04B	RCP M-RCE-P01D (C.E. NO. 2B)
E-NGN-L03C2	FUSE IN BKR.	CTMT NOR DUCT HTR M-HCN-E01C
E-NGN-L03C3	FUSE IN BKR.	CTMT NOR DUCT HTR M-HCN-E01D
E-NGN-L03D2	FUSE IN BKR.	CTMT POLAR CRANE M-ZCN-G01
E-NGN-L06C2	E-NGN-B06C2 (FUSE)	CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01A
E-NGN-L09C4	E-NGN-B09C4 (FUSE)	CTMT PRE-ACCESS NORM AFU FAN M-HCN-F01B
E-NGN-L10C2	FUSE IN BKR.	CTMT NORM DUCT HTR M-HCN-E01A
E-NGN-L10C3	FUSE IN BKR.	CTMT NORM DUCT HTR M-HCN-E01B
J-RCN-PC100A ~~E-NGN-L11C4~~ (Fuse)	C4 E-NGN-L11B2	PROPORTIONAL HTR BANK M-RCE-B2, B8, B14
J-RCN-PC100B ~~E-NGN-L12C4~~ (Fuse)	C4 E-NGN-L12B2	PROPORTIONAL HTR BANK M-RCE-B5, B11, B17
CEA 06 CB101	F101, F102, F103	CEA 06
CEA 08 CB102	F104, F105, F106	CEA 08
CEA 10 CB103	F107, F108, F109	CEA 10

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D13-22	E-NHN-M1329	STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01AH
E-NHN-D15-01	E-NHN-M1526	REACTOR COOLANT PUMP MOTOR SPACE HEATER ~~CONTACTOR~~ M-RCE-P01B H
E-NHN-D15-02	E-NHN-M1526	REACTOR COOLANT PUMP MOTOR SPACE HEATER ~~CONTACTOR~~ M-RCE-P01D1 H
E-NHN-D15-06	E-NHN-M1526	CONTAINMENT PREACCESS NORMAL AFU FAN MOTOR SPACE METER H A M-HCN-F01E BH
E-NHN-D10-01	E-NHN-M1027	REACTOR COOLANT PUMP MOTOR SPACE HEATER ~~CONTACTOR~~ M-RCE-P01AH
E-NHN-D10-02	E-NHN-M1027	REACTOR COOLANT PUMP MOTOR SPACE HEATER ~~CONTACTOR~~ M-RCE-P01CH
E-NHN-D10-20	E-NHN-M1027	STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01B H
E-NHN-D19-05	E-NHN-M1914	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02AH
E-NHN-D19-06	E-NHN-M1914	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02CH
E-NHN-D19-07	E-NHN-M1914	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01AH
E-NHN-D19-08	E-NHN-M1914	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01CH
E-NHN-D19-10	E-NHN-M1914	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03AH
E-NHN-D19-12	E-NHN-M1914	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03CH

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TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-NHN-D20-05	E-NHN-M2013	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02BH
E-NHN-D20-06	E-NHN-M2013	CEDM NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A02DH
E-NHN-D20-07	E-NHN-M2013	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01D H
E-NHN-D20-08	E-NHN-M2013	CONTAINMENT NORMAL ACU FAN MOTOR SPACE HEATER M-HCN-A01BH
E-NHN-D20-10	E-NHN-M2013	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03BH
E-NHN-D20-12	E-NHN-M2013	REACTOR CAVITY NORMAL COOLING FAN MOTOR SPACE HEATER M-HCN-A03DH

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-UV-622
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TANK VENT VALVE J-SIB-HV-613
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TANK VENT VALVE J-SIB-HV-623
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TANK VENT VALVE J-SIB-HV-633
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ TANK VENT VALVE J-SIB-HV-643
E-ZJA-C01 (FUSE)	E-PKA-D2101	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-639
E-ZJA-C01 (FUSE)	E-PKA-D2101	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-649
E-ZJA-C03 (FUSE)	E-PKA-D2111	RCP CONTROLLED BLEEDOFF TO RDT VALVE J-CHA-HV-507
E-ZJA-C03 (FUSE)	E-PKA-D2111	LETDOWN LINE TO REGEN HEAT EXCH CTMT ISO VALVE J-CHA-HV-516
E-ZJA-C03 (FUSE)	E-PKA-D2111	RCP CONTROLLED BLEEDOFF TO VCT VALVE J-CHA-UV-506
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-641
E-ZJB-C01 (FUSE)	E-PKB-D2201	SI TANK CHECK VALVE LEAKAGE ISO VALVE J-SIB-UV-648
E-ZJB-C01 (FUSE)	E-PKB-D2201	HOT LEG INJECT CHECK VLV LEAKAGE ISO VLV J-SIB-UV-322
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-UV-632
E-ZAB-C06 (FUSE)	E-PKB-D2221	REACTOR COOLANT VENT VALVE J-RCB-HV-105
E-ZAB-C06 (FUSE)	E-PKB-D2221	SAFETY INJ. TANK NITROGEN SUPPLY VALVE J-SIB-UV-612

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION	
E-ZJB-C01 (FUSE)	E-PKB-D2201	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-UV-642	
E-ZJB-C03 (FUSE)	E-PKB-D2211	LETDOWN LINE TO REGEN HEAT EXCH VALVE J-CHB-UV-515	
E-ZJB-C03 (FUSE)	E-PKB-D2211	SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-631	
E-ZAA-C03 (FUSE)	E-PKA-D2109	REACTOR DRAIN TANK OUTLET ISOLATION VALVE J-CHAUV-560	
E-ZAA-C03 (FUSE)	E-PKA-D2109	SI TANK RWT HDR CTMT ISOLATION VALVE J-SIA-UV-682	
E-ZAA-C03 (FUSE)	E-PKA-D2109	REGENERATIVE HEAT EXCH TO AUX SPRAY VALVE J-CHA-HV-209 5	X
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISOLATION VALVE J-SSA-UV-203	
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISOLATION VALVE J-SSA-UV-204	
E-ZAA-C01 (FUSE)	E-PKA-D2110	SAMPLE CONTAINMENT ISOLATION VALVE J-SSA-UV-205	
E-ZAA-C04 (FUSE)	E-PKA-D2102	PRESSURIZER VENT VALVE J-RCA-HV-103	
E-ZAA-C04 (FUSE)	E-PKA-D2130	CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-4B	X
E-ZAA-C05 (FUSE)	E-PKA-D2114	STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGA-UV-500P	
E-ZAA-C04 (FUSE)	E-PKA-D2130	CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-4A	X
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-204	
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-211	
E-ZAA-C05 (FUSE)	E-PKA-D2114	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGA-UV-220	
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-619	
E-ZJB-C03 (FUSE)	E-PKB-D2211	SI TANK CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-638	X
E-ZJB-C03 (FUSE)	E-PKB-D2211	HOT LEG INJ CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-332	X
E-ZAA-C03 (FUSE)	E-PKA-D2109	REACTOR COOLANT VENT VALVE J-RCA-HV-101	X

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIA-HV-629
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-605
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-606
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-607
E-ZAA-C06 (FUSE)	E-PKA-D2121	SAFETY INJ TANK VENT VALVE J-SIA-HV-608
E-ZAA-C06 (FUSE)	E-PKA-D2121	RC SYSTEM VENT TO CTMT VALVE J-RCA-HV-106
E-ZAB-C03 (FUSE)	E-PKB-D2209	REGEN HEAT EXCH TO AUX SPRAY VALVE J-CHB-HV-203
E-ZAB-C03 (FUSE)	E-PKB-D2209	REACTOR COOLANT VENT VALVE J-RCB-HV-102
E-ZAB-C03 (FUSE)	E-PKB-D2209	SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-611
E-ZAB-C03 (FUSE)	E-PKB-D2209	SI TANK CHECK VALVE LEAKAGE LINE ISO VALVE J-SIB-UV-618
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADIATION MONITORING ISO VALVE J-HCB-UV-44
E-ZAB-C01 (FUSE)	E-PKB-D2210	CTMT ATM RADIATION MONITORING ISO VALVE J-HCB-UV-47
E-ZAB-C04 (FUSE)	E-PKB-D2202	REACTOR COOLANT VENT VALVE J-RCB-HV-108
E-ZAB-C04 (FUSE)	E-PKB-D2202	SAFETY INJ TANK FILL AND DRAIN VALVE J-SIB-UV-621
E-ZAB-C04 (FUSE)	E-PKB-D2202	SI TANK CHECK VALVE LEAKAGE LINE ISO VALVE J-SIB-UV-628
E-ZAB-C01 (FUSE)	E-PKB-D2210	CONTAINMENT POWER ACCESS PURGE MODE ISOLATION VALVE J-CPB-UV-5A
E-ZAB-C01 (FUSE)	E-PKB-D2210	CONTAINMENT POWER ACCESS PURGE MODE ISOLATION VALVE J-CPB-UV-5B

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION	
E-ZAB-C05 (FUSE)	E-PKB-D2214	REACTOR COOLANT VENT VALVE J-RCB-HV-109	
E-ZAB-C05 (FUSE)	E-PKB-D2214	STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGB-UV-500R	
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-222	
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-224	
E-ZAB-C05 (FUSE)	E-PKB-D2214	BLOWDOWN SAMPLE CTMT ISOLATION VALVE J-SGB-UV-226	
~~E-ZAB-C06 (FUSE)~~	~~E-PKB-D2221~~	~~REACTOR COOLANT VENT VALVE J-RCB-HV-105~~	X
~~E-ZAB-C06 (FUSE)~~	~~E-PKB-D2221~~	~~SAFETY INJ TANK NITROGEN SUPPLY VALVE J-SIB-UV-612~~	X
~~E-ZJB-C03 (FUSE)~~	~~E-PKB-D2211~~	~~SI TANK CHECK VLV LEAKAGE LINE ISO VALVE J-SIB-UV-638~~	X
~~E-ZJB-C03 (FUSE)~~	~~E-PKB-D2211~~	~~HOT LEG INJECT CHECK VLV LEAKAGE ISO VLV J-SIB-UV-332~~	X
E-ZAN-C01 (FUSE)	E-NKN-D4226	SEAL INJECT VALVES TO RCP J-CHE-FV-241	
E-ZAN-C01 (FUSE)	E-NKN-D4224	SEAL INJECT VALVES TO RCP J-CHE-FV-242	
E-ZAN-C01 (FUSE)	E-NKN-D4222	SEAL INJECT VALVES TO RCP J-CHE-FV-244	
E-ZAN-C01 (FUSE)	E-NKN-D4224	POST ACDT SMPLG SYS ISO VALVE J-CHN-HV-923	
E-ZAN-C01 (FUSE)	E-NKN-D4224	REACTOR VESSEL SEAL DRAIN TO RDT VALVE J-RCE-HV-403	
E-ZAN-C01 (FUSE)	E-NKN-D4224	SI DRAIN TO REACTOR DRAIN TANK VALVE J-SIE-HV-661	

TABLE 3.8-3

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIA-UV-647	HPSI A Flow Control to Reactor Coolant Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-637	HPSI A Flow Control to Reactor Coolant Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-604	HPSI Pump A Long Term Cooling Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-609	HPSI Pump B Long Term Cooling Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-657	Shutdown Clg. Temp. Control Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-658	Shutdown Clg. Temp. Control Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-685	LPSI - Ctmt Spray Pump Cross Connect A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-694	LPSI- Ctmt Spray Pump Cross Connect B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-686	Ctmt Spray A Cross Connect Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-696	Ctmt Spray B Cross Connect Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-688	Shutdown Clg. Heat Exchange A Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-693	Shutdown Clg. Heat Exchange B Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-617	HPSI A Flow Control To React Coolant 2A Valve	Safety Injection Shutdown Clg. Sys.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIA-UV-627	HPSI A Flow Control To React Coolant 2B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-645	LPSI Flow Control To React Coolant 1B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-635	LPSI Flow Control To React Coolant 1A Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-644	Safety Injection Tank 1B Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-634	Safety Injection Tank 1A Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-616	HPSI B Flow Control To React Coolant 2A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-626	HPSI B Flow Control To React Coolant 2B Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-636	HPSI B Flow Control To React Coolant 1A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-646	HPSI B Flow Control To React Coolant 1B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-655	Shutdown Clg. Ctmt Isolation Loop 1 Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-656	Shutdown Clg. Ctmt Isolation Loop 2 Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-664	Ctmt Spray Pump A To Refueling Water Tank Isolation Vlv.	Safety Injection Shutdown Clg. Sys.

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TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIB-UV-665	Ctmt Spray Pump B To Refueling Water Tank Isolation Vlv.	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-615	LPSI Flow Control To React Coolant 2A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-625	LPSI B Flow Control To React Coolant 2B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-666	HPSI Pump A to Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-667	HPSI Pump B to Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-669	LPSI Pump A To Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-668	LPSI Pump B to Refueling Water Tank Isolation	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-672	Ctmt Spray Control Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-671	Ctmt Spray Control Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-674	Ctmt Sump Isolation Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-676	Ctmt Sump Isolation Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-UV-651	Shutdown Clg. Isolation Loop 1 Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-652	Shutdown Clg. Isolation Loop 2 Valve	Safety Injection Shutdown Clg. Sys.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIA-UV-673	Ctmt Sump Isolation Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-675	Ctmt Sump Isolation Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-614	Safety Injection Tank 2A Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-UV-624	Safety Injection Tank 2B Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-684	Shutdown Clg. Heat Exchange Isolation Train A	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-689	Shutdown Clg. Heat Exchange Isolation Train B	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-683	LPSI Pump A Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-692	LPSI Pump B Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-691	Shutdown Clg. Loop 2 Warm-Up Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-690	Shutdown Clg. Loop 1 Warm-Up Bypass Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-698	HPSI Pump A Discharge Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-699	HPSI Pump B Discharge Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-306	LPSI Pump A Header Discharge Valve	Safety Injection Shutdown Clg. Sys.

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TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-SIB-HV-307	LPSI Pump B Header Discharge Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-687	Ctmt Spray Isolation Train A Valve	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-695	Ctmt Spray Isolation Train B Valve	Safety Injection Shutdown Clg. Sys.
J-SIA-HV-678	Shutdown Clg. Heat Exchange Isolation Train A	Safety Injection Shutdown Clg. Sys.
J-SIB-HV-679	Shutdown Clg. Heat Exchange Isolation Train B	Safety Injection Shutdown Clg. Sys.
J-SIC-UV-653	Shutdown Clg. Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-SID-UV-654	Shutdown Clg. Isolation Valve	Safety Injection Shutdown Clg. Sys.
J-EWA-UV-65	ECW Loop A To/From NCW Cross Tie Valve	Essential Cooling Water System
J-EWA-UV-145	ECW Loop A To/From NCW Cross Tie Valve	Essential Cooling Water System
J-CTA-HV-1	Condensate Tank to Aux. Feedwater Pump Valve	Condensate Transfer & Storage Sys.
J-CTA-HV-4	Condensate Tank to Aux. Feedwater Pump Valve	Condensate Transfer & Storage Sys.
J-SGA-UV-134	SG-1 Aux. Feedwater Pump A Steam Supply	Main Steam System
J-SGA-UV-138	SG-2 Aux. Feedwater Pump A Steam Supply	Main Steam System

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-NCB-UV-401	NCWS Ctmt Isolation Valve	Nuclear Cooling Water System
J-NCA-UV-402	NCWS Ctmt Isolation Valve	Nuclear Cooling Water System
J-NCB-UV-403	NCWS Ctmt Isolation Valve	Nuclear Cooling Water System
J-AFB-HV-30	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFB-HV-31	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFB-UV-34	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFB-UV-35	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFA-HV-32	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-AFA-UV-37	Aux. Feedwater Isolation Valve	Auxiliary Feed-water System
J-AFC-UV-36	Aux. Feedwater Isolation Valve	Auxiliary Feed-water System
J-AFC-HV-33	Aux. Feedwater Regulating Valve	Auxiliary Feed-water System
J-CPA-UV-2A	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System
J-CPB-UV-3B	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System
J-CPA-UV-2B	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE ~~(Continuous)~~ (Accident Conditions)	SYSTEM(S) AFFECTED
J-CPB-UV-3A	Ctmt Purge Refueling Mode Isolation Valve	Containment Purge System
J-WCA-UV-62	Normal Chill Water Return Ctmt Isolation	Chilled Water System
J-WCB-UV-63	Normal Chill Water Supply Ctmt Isolation	Chilled Water System
J-WCB-UV-61	Normal Chill Water Return Ctmt Isolation	Chilled Water System
J-RDA-UV-23	Ctmt Radwaste Sumps Internal Isolation	Radioactive Waste Drain System
J-HPA-UV-3	H_2 Ctmt Train A Downstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPA-UV-5	H_2 Ctmt Train A Return Isolation Valve	Containment Hydrogen Control Sys.
J-HPB-UV-4	H_2 Ctmt Train B Downstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPB-UV-6	H_2 Ctmt Train B Return Isolation Valve	Containment Hydrogen Control Sys.
J-HPB-UV-2	H_2 Ctmt Train B Upstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPA-UV-1	H_2 Ctmt Train A Upstream Supply Isolation	Containment Hydrogen Control Sys.
J-GRA-UV-1	Radioactive Drain Tk Gas Surge Hdr Internal Containment Isolation	Gaseous Radwaste System

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (Continuous) (Accident Conditions)	SYSTEM(S) AFFECTED
J-HPB-UV-2	H_2 Ctmt Train B Upstream Supply Isolation	Containment Hydrogen Control Sys.
J-HPA-UV-1	H_2-Ctmt Train A Upstream Supply Isolation	Containment Hydrogen Control Sys.
J-GRA-UV-1	Radioactive Drain Tk Gas Surge Hdr Internal Containment Isolation	Gaseous Radwaste System

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3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. ~~The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974.~~ 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, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is 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 required steady state frequency for the emergency diesels is 60 + 1.2/ -0.3 Hz to be consistent with the safety analysis to provide adequate safety injection flow.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977. ~~and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.~~

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A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

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

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

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.010 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

If any other metallic structures (e.g., buildings, new or modified piping systems, conduit) are placed in the ground in the vicinity of the fuel oil storage system or if the original system is modified, the adequacy and frequency of inspections of the cathodic protection system shall be re-evaluated and adjusted in accordance with Regulatory Guide 1.137.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

startup channel

3.9.2 As a minimum, two source-range neutron flux monitors shall be OPERABLE and operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

startup channel

4.9.2 Each source-range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLF and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps.

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LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 4000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs or during surveillance testing of ECCS pumps.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

With the requirement of the specification 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 limit within 4 hours. The provisions of Specification 3.0.3 are not applicable. X

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2150 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

startup channel

The OPERABILITY of the source-range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion, and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, and the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.~~3~~ 2 and 3.3.3.2 during PHYSICS TESTS above 20% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirement of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended. X

TABLE 4.11-2 (Continued)

TABLE NOTATION

[a]The LLD is the smallest concentration of radioactive material in a sample that will yield a net count above background that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume). Current literature defines the LLD as the detection capability for the instrumentation only and the MDC minimum detectable concentration, as the detection capability for a given instrument procedure and type of sample.

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22 is the number of transformations per minute per picocurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between the midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.*

*For a more complete discussion of the LLD, and other detection limits, see the following:
(1) HASL Procedures Manual, HASL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry" Anal. Chem. 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlantic Richfield Hanford Company Report (ARH-2537 (June 22, 1972).

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TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY[a]	TYPE AND FREQUENCY OF ANALYSIS[e]
Waterborne			
Surface	Water storage reservoir (#60) evaporation pond (#59)	Monthly composite of weekly grab sample	Gamma isotopic analysis monthly; tritium quarterly
Ground	2 onsite wells[g] (#57, 58)	Quarterly grab sample	Tritium and gamma isotopic analysis quarterly
Drinking (well)	3 wells from surrounding residences (#46, 48, 49) that would be affected by its discharge	Composite sample of weekly grab samples over 2-week period when I-131 analysis is performed, monthly composite of weekly grab samples otherwise	I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.[h] Composite for gross beta and gamma isotopic analyses monthly. Composite for tritium analysis quarterly.
Ingestion			
Milk	Samples from milking animals in 3 locations ~~(#50, 51, 53)~~ within 5 km distance having the highest dose potential. If there are none, 1 sample from milking animals in each of 3 areas (#50, 51, 53) between 5 and 8 km distant where doses are calculated to be greater than 1 mrem per year.[h] One sample from milking animals at a control location (#56), 15 to 30 km distant and in the least prevalent wind direction.	Semimonthly for animals on pasture; otherwise, monthly	Gamma isotopic and I-131 analysis semi-monthly when animals are on pasture or monthly at other times x

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[a]	SAMPLING AND COLLECTION FREQUENCY[a]	TYPE AND FREQUENCY OF ANALYSIS[e]
Food products*	Samples (#14, 46, #47, 52) of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed	Monthly during growing season	Gamma isotopic and I-131 analysis.
	1 sample (#51, 62) of each of the similar broad leaf vegetation grown 15-30 km distant in the least prevalent wind direction if milk sampling is not performed	Monthly during growing season	Gamma isotopic and I-131 analysis.

*When broad leaf vegetation samples are not available, reports from 4 existing supplemental airborne radioiodine sample locations will be substituted.

TABLE 4.12-1 (Continued)

TABLE NOTATION

In calculating the LLD for a radionuclide determined by gamma-ray spectrometry the background should include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report.

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UNIT STAFF (Continued)

b. Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a nominal 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modifications, on a temporary basis, the following guidelines shall be followed (this excludes the STA working hours):

   1) An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

   2) An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

   3) A break of at least 8 hours should be allowed between work periods, including shift turnover time.

   4) Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

c. Any deviation from the above guidelines shall be authorized by the PVNGS Plant Manager or his designee who is at supervisory level or above, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the PVNGS Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

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TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, OR 4	MODE 5 OR 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License
SRO - Individual with a Senior Reactor Operators License
RO - Individual with a Reactor Operators License
AO - Nuclear Operator I or II
STA - Shift Technical Advisor

~~Except for the Shift Supervisor,~~ the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3, or 4, an individual ~~(other than the Shift Technical Advisor)~~ with a valid Senior Operator license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator or Operator license shall be designated to assume the Control Room command function.

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QUORUM

6.5.1.5 The quorum of the PRB necessary for the performance of the PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman, Vice-Chairmen, or his designated alternate and five members including alternates.

RESPONSIBILITIES

6.5.1.6 The PRB shall be responsible for:

a. Review of all administrative control procedures and changes.

b. Review of all proposed changes to Appendix "A" Technical Specifications.

c. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Nuclear Safety Group (NSG).

d. Review of REPORTABLE EVENTS. ~~requiring 24-hour written notification to the Commission.~~

e. Review of unit operations to detect potential nuclear safety hazards.

f. Performance of special reviews, investigations or analyses and reports thereon as requested by the PVNGS Plant Manager.

g. Review and documentation ~~or~~ of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last PRB meeting.

AUTHORITY

6.5.1.7 The PRB shall:

a. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6c. above constitutes an unreviewed safety question.

b. Provide written notification within 24 hours to the Vice President-Nuclear Production, PVNGS Plant Manager and NSG of disagreement between the PRB and the PVNGS Plant Manager; however, the PVNGS Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

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RECORDS

6.5.1.8 The PRB shall maintain written minutes of each PRB meeting that, at a minimum, document the results of all PRB activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the PVNGS Plant Manager and NSG.

6.5.2 TECHNICAL REVIEW AND CONTROL ACTIVITIES

ACTIVITIES

6.5.2.1 The PVNGS Plant Manager shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Phase I - IV tests described in the FSAR that are performed by the plant operations staff shall be approved by the Manager of Technical Support or the Manager of Engineering as previously designated by the PVNGS Plant Manager. Test results shall be approved by the PVNGS Plant Manager or the Manager Technical Support.

6.5.2.3 Proposed modifications to unit nuclear safety-related structures, systems and components shall be designed by a qualified individual/organization. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modification. Proposed modifications to nuclear safety-related structures, systems and components shall be approved prior to implementation by the PVNGS Plant Manager; or by the Manager Technical Support as previously designated by the PVNGS Plant Manager.

6.5.2.4 Individuals responsible for reviews performed in accordance with 6.5.2.1, 6.5.2.2, and 6.5.2.3 shall be members of the station supervisory staff, previously designated by the PVNGS Plant Manager to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate designated review personnel.

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the PVNGS Plant Manager, the Manager Technical Support, the Manager Operations, or the Manager Maintenance.

6.5.2.6 The station security program and implementing procedures shall be reviewed. Recommended changes shall be approved by the PVNGS Plant Manager or designated alternate and transmitted to the Vice President-Nuclear Production and to the NSG.

6.5.2.7 The station emergency plan and implementing procedures shall be reviewed. Recommended changes shall be approved by the PVNGS Plant Manager or designate alternate and transmitted to the Vice President-Nuclear Operations and to the NSG.

6.5.2.8 The PVNGS Plant Manager shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering the evaluation, recommendations and disposition of the corrective action to prevent recurrence.

ACTIVITIES

TECHNICAL REVIEW AND CONTROL (Continued)

6.5.2.9 The PVNGS Plant Manager shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, radwaste treatment systems, and the Pre-planned Alternate Sampling Program.

6.5.2.10 Reports documenting each of the activities performed under Specifications 6.5.2.1 through 6.5.2.9 above shall be maintained. Copies shall be provided to the PVNGS Plant Manager and the Nuclear Safety Group.

6.5.3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.3.1 The NSG shall function to provide independent review and shall be responsible for the audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy
e. instrumentation and control
f. radiological safety
g. mechanical and electrical engineering
h. quality assurance practices

COMPOSITION

6.5.3.2 The NSG shall consist of a Supervisor and at least four staff specialists. The supervisor shall have a Bachelor's Degree in Engineering or the Physical Sciences. He will also have a minimum of 10 years experience in the power field with at least 3 of those years in the nuclear field. The NSG Supervisor will have at least 2 years of supervisor/managerial experience. Each staff specialist will have at least one of the following requirements:

a. Eight years experience in one of the designated areas in Specification 6.5.3.1. One of these 8 years will be at Palo Verde Nuclear Generating Station.

b. Bachelor's Degree in Engineering or a related science and 3 years of professional experience.

CONSULTANTS

6.5.3.3 Consultants shall be utilized as determined by the NSG Supervisor to provide expert advice to the NSG.

REVIEW

6.5.3.4 The NSG shall review:

a. The safety evaluations program and its implementation for (1) changes to procedures, equipment, systems or facilities within the power block, and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

AUDITS (Continued)

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months.

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

k. The performance of activities required by the Operations Quality Assurance Criteria Manual to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months.

~~l. The Pre-planned Alternate Sampling Program and implementing procedures at least once per 24 months.~~ X

AUTHORITY

6.5.3.6 The NSG shall report to and advise the Manager of Nuclear Safety on those areas of responsibility specified in Specifications 6.5.3.4 and 6.5.3.5.

RECORDS

6.5.3.7 Records of NSG activities shall be prepared and maintained. Report of reviews and audits shall be prepared monthly for the Manager of Nuclear Safety who will distribute it to the Vice President-Nuclear Production, PVNGS Plant Manager, and to the management positions responsible for the areas audited.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

pursuant to the requirements of Section 50.72 to 10CFR Part 50

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT ~~requiring 24 hours written notification~~ shall be reviewed by the PRB, and the results of this review shall be submitted to the Supervisor of Nuclear Safety Group and the Vice President-Nuclear Production.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President-Nuclear Production, PVNGS Plant Manager and Supervisor of Nuclear Safety Group shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRB. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the Supervisor of the NSG and the Vice President-Nuclear Production within ~~14~~ 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG-0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

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

NOTES (1) Modification to the CPC Addressable Constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the PRB. (2) ... (see insert)

h. PROCESS CONTROL PROGRAM implementation.

i. OFFSITE DOSE CALCULATION MANUAL implementation.

j. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

k. Pre-planned Alternate Sampling Program implementation.

l. Secondary water chemistry program implementation.

(Insert)

(2) Modifications to the CPC software (including algorithm changes and changes in fuel cycle specific data) shall be performed in accordance with the most recent version of CEN-39(A)-P, "CPC Protection Algorithm Software Change Procedure", that has been determined to be applicable to the facility. Additions or deletions to CPC Addressable Constants or changes to Addressable Constant software limit values shall not be implemented without prior NRC approval.

PROCEDURES AND PROGRAMS (Continued)

(1) Training of personnel, and

(2) Procedures for monitoring.

e. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis,

(3) Provisions for maintenance of sampling and analysis equipment.

f. Spray Pond Monitoring

A program which will identify and describe the parameters and activities used to control and monitor the Essential Spray Pond and Piping. The program shall be conducted in accordance with station ~~Procedure 73AG-SP01~~. manual procedures.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The Startup Report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

add attachment 1

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly ~~basis to the Director,~~ Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT~~***~~ *

6.9.1.7 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

**This tabulation supplements the requirements of §20.407 of the 10 CFR Part 20.

~~***A single submittal may be made for a multiple unit station.~~

ATTACHMENT 1

Annual reports shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.7. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL.

The Semiannual Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

The Semiannual Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

*In lieu of submission with the first half year Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made; and

7) An estimate of the exposure to plant operating personnel as a result of the change.

~~6.16 PRE-PLANNED ALTERNATE SAMPLING PROGRAM (PASP)~~

~~6.16.1 The PASP shall be approved by the Regional Administrator, U.S. NRC Region V, prior to implementation.~~

~~6.16.2 Licensee-initiated changes to the PASP:~~

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

~~1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information; and~~

~~2) A determination that the change did not reduce the overall effectiveness of the Gaseous Effluent Sampling Program.~~

~~3) A determination that the change did not reduce the overall effectiveness of the Post Accident Sampling System.~~

~~4) A determination that the change did not reduce the overall effectiveness of the post-accident High Range Effluent Monitors.~~

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. ~~The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical~~ Power Sources," December 1974. 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, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is 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 required steady state frequency for the emergency diesels is 60 + 1.2/-0.3 Hz to be consistent with the safety analysis to provide adequate safety injection flow.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The surveillance requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 ~~and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.~~

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