March 2, 1988

Docket No.: STN 50-528

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Mr. E. E. Van Brunt, Jr.
Executive Vice President
Arizona Nuclear Power Project
Post Office Box 52034
Phoenix, Arizona 85072-2034

Dear Mr. Van Brunt:

SUBJECT: ISSUANCE OF AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF-41 FOR THE PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1 (TAC NO. 63010)

The Commission has issued the subject Amendment, which is enclosed, to the Facility Operating License for Palo Verde Nuclear Generating Station, Unit 1. The Amendment consists of changes to the Technical Specifications (Appendix A to the license) in response to your application transmitted by letter dated May 25, 1987, as amended by letter dated August 7, 1987.

The Amendment revises the Technical Specifications in a number of areas in order to be consistent with those areas of the Technical Specifications for Palo Verde, Units 2 and 3, previously reviewed and approved by the staff.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commissions next regular bi-weekly Federal Register notice.

Sincerely,

E. A. Licitra, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 27 to NPF-41
2. Safety Evaluation

cc: See next page.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

DOCKET NO. STN 50-528

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. NPF-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment, dated May 25, 1987 as amended by letter dated August 7, 1987, by the Arizona Public Service Company (APS) on behalf of itself and the Salt River Project Agricultural Improvement and Power District, El Paso Electric Company, Southern California Edison Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the enclosure to this license amendment, and paragraph 2.C(2) of Facility Operating License No. NPF-41 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 27, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. APS shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license Amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert B. Samworth

For George Knighton, Director
Project Directorate V
Division of Reactor Projects - III, IV, V and Special Projects

Enclosure:
Changes to the Technical Specifications

Date of Issuance: March 2, 1988

March 2, 1988

ENCLOSURE TO LICENSE AMENDMENT

AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF-41

DOCKET NO. STN 50-528

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change. Also to be replaced are the following overleaf pages to the amended pages.

Amendment Pages	Overleaf Pages
V	-
VI	-
VIII	VII
XI	-
XII	-
XVII thru XX	-
XXI	XXII
1-5	1-6
3/4 1-4	3/4 1-3
3/4 1-14	3/4 1-13
3/4 1/21	3/4 1-22
3/4 1-25	3/4 1-26
B 3/4 1-2	B 3/4 1-1
3/4 2-2	3/4 2-1
3/4 2-3	3/4 2-4
3/4 2-9	3/4 2-10
3/4 2-12	3/4 2-11
3/4 3-16	3/4 3-15
3/4 3-22	3/4 3-21
3/4 3-23	-
3/4 3-24	-
3/4 3-29	3/4 3-30
3/4 3-36	3/4 3-35
3/4 3-39	3/4 3-40
3/4 3-48	3/4 3-47
3/4 3-49 thru 3-54	-
3/4 3-55	3/4 3-56
3/4 3-58	3/4 3-57

Amendment Pages	Overleaf Pages
3/4 3-59	-
3/4 3-60	-
3/4 3-65	3/4 3-66
3/4 3-67	-
3/4 3-68	-
3/4 3-70	3/4 3-69
3/4 3-72	3/4 3-71
B 3/4 3-1	B 3/4 3-2
B 3/4 3-3	B 3/4 3-4
3/4 4-1	-
3/4 4-2	-
3/4 4-3	3/4 4-4
3/4 4-5 thru 4-8	-
3/4 4-10	3/4 4-9
3/4 4-16	3/4 4-15
3/4 4-19	-
3/4 4-20	-
3/4 4-23	3/4 4-24
3/4 4-25 thru 4-32	-
3/4 4-33	3/4 4-34
3/4 4-35	-
B 3/4 4-5	-
B 3/4 4-6	-
B 3/4 4-7	B 3/4 4-8
B 3/4 4-10	B 3/4 4-9
B 3/4 4-11	B 3/4 4-12
3/4 5-1	3/4 5-2
3/4 5-5	-
3/4 5-6	-
B 3/4 5-3	B 3/4 5-4

Amendment Pages	Overleaf Pages
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3/4 6-5	3/4 6-6
3/4 6-7	3/4 6-8
3/4 6-15	3/4 6-16
3/4 6-17	3/4 6-18
3/4 6-19	3/4 6-20
3/4 6-22	3/4 6-21
3/4 6-35	-
3/4 6-36	-
3/4 6-38	3/4 6-37
B 3/4 6-2	B 3/4 6-1
B 3/4 6-4	B 3/4 6-3
3/4 7-1	3/4 7-2
3/4 7-6	3/4 7-5
3/4 7-10	3/4 7-9
3/4 7-16	3/4 7-15
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3/4 7-22	-
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3/4 8-24	3/4 8-23
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3/4 8-33 thru 8-38*	-
3/4 8-39	3/4 8-40
3/4 8-41 thru 8-48	-
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B 3/4 8-2	-
3/4 9-2	3/4 9-1
3/4 9-8	3/4 9-7
3/4 9-9	3/4 9-10
3/4 9-13	3/4 9-14
B 3/4 9-1	B 3/4 9-2

*3/4 8-38 has been deleted.

Amendment Pages	Overleaf Pages
3/4 10-2	3/4 10-1
3/4 11-9	3/4 11-10
3/4 12-4	3/4 12-3
3/4 12-5	3/4 12-6
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PRESSURE BOUNDARY LEAKAGE

1.23 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.24 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

1.25 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

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.27 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

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

SHUTDOWN MARGIN

1.29 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.

SITE BOUNDARY

1.30 The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

SOFTWARE

1.31 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation, and procedures.

SOLIDIFICATION

1.32 SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

SOURCE CHECK

1.33 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.34 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

SURVEILLANCE REQUIREMENTS (Continued)

b. When in MODE 1 or MODE 2 with k_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with k_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

e. When in MODE 3, 4, or 5, at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.2.2 When in MODE 3, 4, or 5, with any full-length CEA fully or partially withdrawn, and T_{cold} less than or equal to 500°F, K_{N-1} shall be determined to be less than 0.99 at least once per 24 hours by consideration of at least the following factors:

1. Reactor Coolant System boron concentration,
2. CEA position,
3. Reactor Coolant System average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration

4.1.1.2.3 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.2.1.e or 4.1.1.2.2. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

REACTIVITY CONTROL SYSTEMS

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.

*With K_{eff} greater than or equal to 1.0.

#See Special Test Exception 3.10.2.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 Each of the following borated water sources shall be OPERABLE:

a. The spent fuel pool with:

1. A minimum borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 ppm and 4400 ppm boron, and
3. A solution temperature between 60°F and 180°F.

b. The refueling water tank with:

1. A minimum contained borated water volume as specified in Figure 3.1-2, and
2. A boron concentration of between 4000 and 4400 ppm of boron, and
3. A solution temperature between 60°F and 120°F.

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

ACTION:

a. With the above required spent fuel pool inoperable, restore the pool to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the above required spent fuel pool to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each of the above required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water, and
2. Verifying the contained borated water volume of the water source.

b. At least once per 24 hours by verifying the refueling water tank temperature when the outside air temperature is outside the 60°F to 120°F range.

c. At least once per 24 hours by verifying the spent fuel pool temperature when irradiated fuel is present in the pool.

*See Special Test Exception 3.10.7.

REACTIVITY CONTROL SYSTEMS

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.

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.

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 3.1.1.2 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 3.1-2A 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 3.1.1.2 is satisfied. After declaring the CEA(s) inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specifications 3.1.3.6 and 3.1.3.7 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 3.1-2A, 3.1-3 and 3.1-4; the THERMAL POWER level shall be restricted pursuant to Specifications 3.1.3.6 and 3.1.3.7 during subsequent operation.

*See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.2 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 and the CEA is maintained pursuant to the requirements of Specification 3.1.3.7.

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, 3.1.3.6 and 3.1.3.7. 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.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part-length CEA not fully inserted.

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

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The above required CEA Reed Switch Position Transmitter indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip breakers in the closed position.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN and K_{N-1}

The function of SHUTDOWN MARGIN is to ensure that the reactor remains subcritical following a design basis accident or anticipated operational occurrence. The function of K_{N-1} is to maintain sufficient subcriticality to preclude inadvertent criticality following ejection of a single control element assembly (CEA). During operation in MODES 1 and 2, with k_{eff} greater than or equal to 1.0, the transient insertion limits of Specification 3.1.3.6 ensure that sufficient SHUTDOWN MARGIN is available.

SHUTDOWN MARGIN is the amount by which the core is subcritical, or would be subcritical immediately following a reactor trip, considering a single malfunction resulting in the highest worth CEA failing to insert. K_{N-1} is a measure of the core's reactivity, considering a single malfunction resulting in the highest worth inserted CEA being ejected.

SHUTDOWN MARGIN requirements vary throughout the core life as a function of fuel depletion and reactor coolant system (RCS) cold leg temperature (T_{cold}). The most restrictive condition occurs at EOL, with T_{cold} at no-load operating temperature, and is associated with a postulated steam line break accident and the resulting uncontrolled RCS cooldown. In the analysis of this accident, the specified SHUTDOWN MARGIN is required to control the reactivity transient and ensure that the fuel performance and offsite dose criteria are satisfied. As (initial) T_{cold} decreases, the potential RCS cooldown and the resulting reactivity transient are less severe and, therefore, the required SHUTDOWN MARGIN also decreases. Below T_{cold} of about 210°F, the inadvertent deboration event becomes limiting with respect to the SHUTDOWN MARGIN requirements. Below 210°F, the specified SHUTDOWN MARGIN ensures that sufficient time for operator actions exists between the initial indication of the deboration and the total loss of shutdown margin. Accordingly, with at least one CEA partially or fully withdrawn, the SHUTDOWN MARGIN requirements are based upon these limiting conditions.

Additional events considered in establishing requirements on SHUTDOWN MARGIN that are not limiting with respect to the Specification limits are single CEA withdrawal and startup of an inactive reactor coolant pump.

K_{N-1} requirements vary with the amount of positive reactivity that would be introduced assuming the CEA with the highest inserted worth ejects from the core. In the analysis of the CEA ejection event, the K_{N-1} requirement ensures that the radially averaged enthalpy acceptance criterion is satisfied, considering power redistribution effects. Above T_{cold} of 500°F, Doppler reactivity feedback is sufficient to preclude the need for a specific K_{N-1} requirement. With all CEAs fully inserted, K_{N-1} and SHUTDOWN MARGIN requirements are equivalent in terms of minimum acceptable core boron concentration.

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, (4) an emergency power supply from OPERABLE diesel generators, and (5) the volume control tank (VCT) outlet valve CH-UV-501, capable of isolating the VCT from the charging pump suction line. 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.

Each system is capable of providing boration equivalent to a SHUTDOWN MARGIN of 4% delta k/k after xenon decay and cooldown to 210°F. Therefore, the boration capacity of either system is more than sufficient to satisfy the SHUTDOWN MARGIN and/or K_{N-1} requirements of the specifications. 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.

Each system is capable of providing boration equivalent to a SHUTDOWN MARGIN of 4% delta k/k. Therefore, the boration capacity of the system required below 210°F is more than sufficient to satisfy the shutdown margin and/or K_{N-1} requirements of the specifications. This condition requires 9,700 gallons of 4000 ppm borated water from either the refueling water tank or the spent fuel pool.

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate limit of 13.5 kW/ft shall be maintained by one of the following methods as applicable:

a. Maintaining COLSS calculated core power less than or equal to the COLSS calculated power operating limit based on linear heat rate (when COLSS is in service); or

b. Maintaining peak linear heat rate within its limit using any operable CPC channel (when COLSS is out of service).

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

ACTION:

With the linear heat rate limit not being maintained as indicated by:

1. COLSS calculated core power exceeding the COLSS calculated core power operating limit based on linear heat rate; or

2. Peak linear heat rate outside its limit using any operable CPC channel (when COLSS is out of service);

within 15 minutes initiate corrective action to reduce the linear neat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within 1 hour, or

b. Reduce THERMAL POWER to less than or equal to 20% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limit when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on any OPERABLE Local Power Density channel, is within its limit.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on linear heat rate.

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

3.2.2 The measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) shall be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) 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^m_{xy} exceeding a corresponding F^c_{xy}, 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^m_{xy}/F^c_{xy} and restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F^m_{xy}/F^c_{xy}) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected PLANAR RADIAL PEAKING FACTORS (F^c_{xy}) used in the COLSS and CPC to a value greater than or equal to the measured PLANAR RADIAL PEAKING FACTORS (F^m_{xy}) 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^m_{xy}) obtained by using the incore detection system, shall be determined to be less than or equal to the PLANAR RADIAL PEAKING FACTORS (F^c_{xy}), 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.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit above 20% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT less than or equal to the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 EFPD to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

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

FIGURE 3.2-3
REACTOR COOLANT COLD LEG TEMPERATURE vs CORE POWER LEVEL

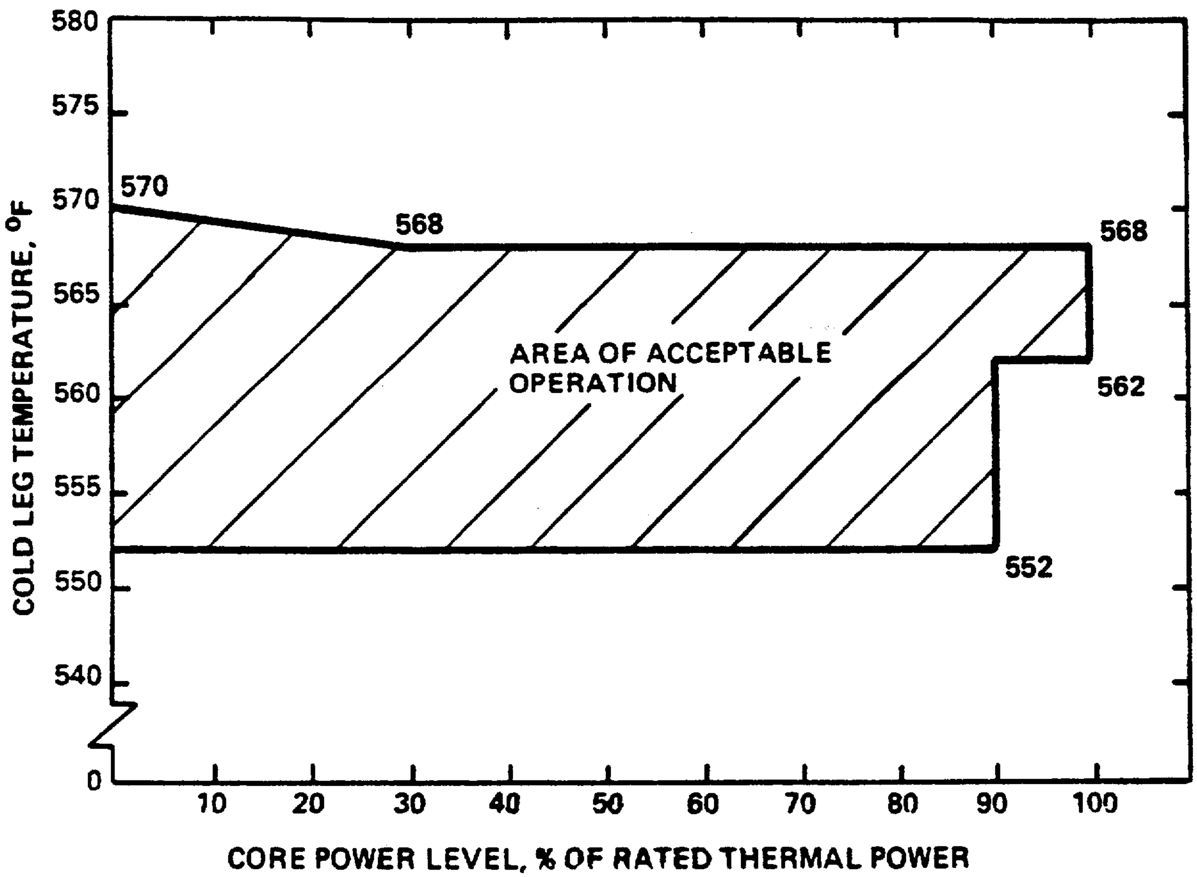

FIGURE 3.2-3

REACTOR COOLANT COLD LEG TEMPERATURE VS CORE POWER LEVEL

3/4.2.7 AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq 0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq + 0.20$

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

ACTION:

With the core average AXIAL SHAPE INDEX outside its above limits, restore the core average ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.7 The core average AXIAL SHAPE INDEX shall be determined to be within its limit at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

3/4.2.8 PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

3.2.8 The pressurizer pressure shall be maintained between 2025 psia and 2300 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 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
D. Supplementary Protection System				
Pressurizer Pressure - High	S	R	M	1, 2
II. RPS LOGIC				
A. Matrix Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
B. Initiation Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*
III. RPS ACTUATION DEVICES				
A. Reactor Trip Breakers	N.A.	N.A.	M, R (10)	1, 2, 3*, 4*, 5*
B. Manual Trip	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

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 subchannel 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.

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

(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.

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
V. RECIRCULATION (RAS)					
A. Sensor/Trip Units					
Refueling Water Storage Tank - Low	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 RAS	4(c)	2(d)	4	1, 2, 3, 4	12
C. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	16
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1)					
A. Sensor/Trip Units					
1. Steam Generator #1 Level - Low	4	2	3	1, 2, 3	13*, 14*
2. Steam Generator Δ Pressure - SG2 > SG1	4	2	3	1, 2, 3	13*, 14*

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*

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

(a) In MODES 3-4, 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 3-4, 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.

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.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 13 are satisfied.

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.

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 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
8. Steam Generator Level - High	
a. Main Steam Isolation	
1. MSIS actuated MSIV's	$\leq$ 5.6*/5.6**
2. MSIS actuated MFIV's#	$\leq$ 10.6*/10.6**
9. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Auxiliary Feedwater Isolation from the Ruptured Steam Generator	$\leq$ 16*/16**
10. Control Room Essential Filtration Actuation	$\leq$ 180*/$\leq$ 180**
11. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	
Loss of Power 90% system voltage	$\leq$ 35.0
12. 4.16 kV Emergency Bus Undervoltage (loss of Voltage)	
Loss of Power	$\leq$ 2.4

TABLE NOTATIONS

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

#MFIV valves tested at simulated operating conditions; valves tested at static flow conditions to $\leq$ 8.6/8.6 seconds.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ESFA SYSTEM FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
VI. AUXILIARY FEEDWATER (SG-1)(AFAS-1) (Continued)				
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
VII. AUXILIARY FEEDWATER (SG-2)(AFAS-2)				
A. Sensor/Trip Units				
1. Steam Generator #2 Level - Low	S	R	M	1, 2, 3
2. Steam Generator Δ Pressure SG1 > SG2	S	R	M	1, 2, 3
B. ESFA System Logic				
1. Matrix Logic	NA	NA	M	1, 2, 3, 4
2. Initiation Logic	NA	NA	M	1, 2, 3, 4
3. Manual AFAS	NA	NA	M	1, 2, 3, 4
C. Automatic Actuation Logic	NA	NA	M(1) (2) (3)	1, 2, 3, 4
VIII. LOSS OF POWER (LOV)				
A. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	S	R	R	1, 2, 3, 4
B. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	S	R	R	1, 2, 3, 4

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		TRAIN B	
ESF FUNCTION	ACTUATION DEVICE	ESF FUNCTION	ACTUATION DEVICE
SIAS A	K108	SIAS B	K108
SIAS A	K409	SIAS B	K409
CIAS A	K202	CIAS B	K204
CIAS A	K204	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		

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 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 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.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Area Monitors				
A. Fuel Pool Area RU-31	S	R	M	**
B. New Fuel Area RU-19	S	R	M	*
C. Containment Power Access Purge Exhaust RU-37 & RU-38	P#	R	P###,W##	##
D. Containment RU-148 & RU-149	S	R	M	1,2,3,4
E. Main Steam RU-139 A&B RU-140 A&B	S	R	M	1,2,3,4
2. Process Monitors				
A. Containment Building Atmosphere RU-1				
1) Particulate	S	R	M	1,2,3,4
2) Gaseous	S	R	M	1,2,3,4
B. Control Room Ventilation Intake RU-29 & RU-30	S	R	M	All MODES
3. Post Accident Sampling System	N.A.	R	M***	1,2,3

*With fuel in the storage pool or building.

**With irradiated fuel in the storage pool.

***The functional test should consist of, but not be limited to, a verification of system sampling capabilities.

#If purge is in service for greater than 12 hours, perform once per 12-hour period.

##When purge system is in operation.

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

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 35 feet	D	SA
b. Nominal Elev. 200 feet	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 35 feet	D	SA
b. Nominal Elev. 200 feet	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 35 feet - 200 feet	D	SA

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 shutdown monitoring channels less than required by Table 3.3-9 A-C, 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.

TABLE 3.3-9A

REMOTE SHUTDOWN INSTRUMENTATION

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

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 Atmospheric Dump Valve Solenoid Air Isolation Valves SGD-HY-178B and SGD-HY-178S	RSP
17. SG 2 line 1 Atmospheric Dump Valve Solenoid Air Isolation Valves SGD-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-E01 Breaker Control	PBB-S04G
26. E-PBB-S04J 4.16KV Feeder Breaker to 480V Load Center PGB-L32	PBB-S04J
27. E-PBB-S04H 4.16KV Feeder Breaker to 480V Load Center PGB-L34	PBB-S04H
28. E-PBB-S04N 4.16KV Feeder Breaker to 480V Load Center PGB-L36	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 Supply Breaker to Load Center PGB-L32	PGB-L32B2
32. E-PGB-L34B2-480V Main Supply Breaker to Load Center PGB-L34	PGB-L34B2
33. E-PGB-L36B2-480V Main Supply Breaker to Load Center PGB-L36	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-UV-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 SIB-HV-307	PHB-M3803
51. LPSI Pump "B" Recirc SIB-UV-668	PHB-M3611
52. LPSI Pump "B" Suction from RWT SIB-HV-692	PHB-M3805
53. SD Cooling LPSI Pump "B" Suction SIB-UV-652	PHB-M3611
54. SD Cooling LPSI Pump "B" Suction SIB-UV-654	PKD-B44
55. LPSI Header "B" to RC Loop 2A SIB-UV-615	PHB-M3611
56. LPSI Header "B" to RC Loop 2B SIB-UV-625	PHB-M3640
57. VCT Outlet Isolation CHN-UV-501	NHN-M7208
58. RWT Gravity Feed CHN-HV-536	NHN-M7209
59. Shutdown Cooling Temperature Control SIB-UV-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	SSA-J04
68. R.C.S. Sample Isolation Valve SSB-UV-200	RSP
69. 125 VDC Battery A Breaker Control Room Circuits	PKA-M4101

TABLE 3.3-9C
REMOTE SHUTDOWN CONTROL CIRCUITS

CONTROL CIRCUITS	SWITCH LOCATION
1. Auxiliary Feedwater Pump B to S/G 1 Isolation Valve AFB-UV-34	RSP
2. Auxiliary Feedwater Pump B to S/G 1 Control Valve AFB-HV-30	RSP
3. Auxiliary Feedwater Pump B to S/G 2 Isolation Valve AFB-UV-35	RSP
4. Auxiliary Feedwater Pump B to S/G 2 Control Valve AFB-HV-31	RSP
5. Auxiliary Feedwater Pump AFB-P01	PBB-S04S
6. Charging Pump No. 2 CHB-P01	PGB-L32C4
7. Pressurizer Auxiliary Spray Valve CHB-HV-203	RSP
8. Pressurizer Backup Heater Bank	RSP
9. Letdown to Regen HX Isolation Valve CHB-UV-515	RSP
10. RCP Cont Bleedoff Valve CHB-UV-505	RSP
11. Volume Control Tank Outlet Isolation Valve CHN-UV-501	NHN-M7208
12. RWT Gravity Feed Isolation Valve CHE-HV-536	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
22. E-PBB-S04H 4.16 KV Feeder Breaker to 480V Load Center PGB-L34 Supply Breaker	PBB-S04H
23. E-PBB-S04J 4.16KV Feeder Breaker to 480V Load Center PGB-L32 Supply Breaker	PBB-S04J
24. E-PBB-S04N 4.16KV Feeder Breaker to 480V Load Center PGB-L36 Supply Breaker	PBB-S04N
25. E-PGB-L32B2 480V Main Supply Breaker to Load Center PGB-L32	PGB-L32B1
26. E-PGB-L34B2-480V Main Supply Breaker to Load Center PGB-L34	PGB-L34B1

TABLE 3.3-9C (continued)
REMOTE SHUTDOWN CONTROL CIRCUITS

	CONTROL CIRCUITS	SWITCH LOCATION
27.	E-PGB-L36 480V Supply Breaker to Load Center PGB-L36	PGB-L36B1
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-HV-613	RSP
40.	SIT 2B Vent Valve SIB-HV-623	RSP
41.	SIT 1A Vent Valve SIB-HV-633	RSP
42.	SIT 1B Vent Valve SIB-HV-643	RSP
43.	LPSI Pump B SIB-P01	PBB-S04F
44.	Containment Spray Pump B Discharger to SD HX "B" Valve SIB-HV-689	PHB-M3804
45.	LPSI Containment Spray from SD HX "B" X-tie Valve SIB-HV-695	PHB-M3810
46.	Shutdown Cooling LPSI Suction Valve SIB-HV-656	PHB-M3605
47.	Shutdown Cooling Warmup Bypass Valve SIB-UV-690	PHB-M3806
48.	LPSI Containment Spray to SD HX "B" X-tie Valve SIB-HV-694	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-HV-696	PHB-M3415
50. LPSI SD HX "B" Bypass Valve SIB-HV-307	PHB-M3803
51. LPSI Pump B Recirc. Valve SIB-UV-688	PHB-M3609
52. LPSI Pump B Suction From RWT SIB-HV-692	PHB-M3805
53. RC Loop to Shutdown Cooling Valve SIB-UV-652	PHB-M3604
54. RC Loop to Shutdown Cooling Valve SIB-UV-654	PKD-B44
55. LPSI Header B to RC Loop 2A Valve SIB-UV-615	PHB-M3606
56. LPSI Header B to RC Loop 2B Valve SIB-UV-625	PHB-M3621
57. SDC "B" Temperature Control Valve SIB-HV-658	PHB-M3412
58. Control Room Ventilation Isolation Dampers HJB-M01/HJB-M55	RSP
59. O.S.A. Supply Damper HJB-M02	RSP
60. O.S.A. Supply Damper HJB-M03	RSP
61. Diesel Generator "B" Emergency Start	DGB-B01
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
67. RCS Sample Isolation Valve SSB-UV-200	SSB-J04
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 4.3-6

REMOTE SHUTDOWN INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Log Neutron Power Level	M	R
2. Reactor Coolant Hot Leg Temperature (2)	M	R
3. Reactor Coolant Cold Leg Temperature (2)	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level	M	R
8. Refueling Water Tank Level	M	R
9. Charging Line Pressure	M	R
10. Charging Line Flow	M	R
11. Shutdown Cooling Heat Exchanger Temperatures	M	R
12. Shutdown Cooling Flow	M	R
13. Auxiliary Feedwater Flow Rate	M	R

POST-ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more accident monitoring instrumentation channels inoperable, take the action shown in Table 3.3-10.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

POST-ACCIDENT 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.

TABLE 3.3-10
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 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

TABLE 3.3-12 (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

TABLE 3.3-12 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. PLANT VENT SYSTEM (Continued)			
B. High Range Monitors			
a. Noble Gas Activity Monitor #RU-144	1	*	42
b. Iodine Sampler	1	*	42
c. Particulate Sampler	1	*	42
d. Sampler Flow Rate Measuring Device	1	*	42
5. FUEL BUILDING VENTILATION SYSTEM			
A. Low Range Monitors			
a. Noble Gas Activity Monitor #RU-145	1	##	37,41
b. Iodine Sampler	1	##	40
c. Particulate Sampler	1	##	40
d. Flow Rate Monitor	1	##	36
e. Sampler Flow Rate Measuring Device	1	##	36
B. High Range Monitors			
a. Noble Gas Activity Monitor #RU-146	1	##	41,42
b. Iodine Sampler	1	##	42
c. Particulate Sampler	1	##	42
d. Sampler Flow Rate Measuring Device	1	##	42

TABLE 3.3-12 (Continued)

TABLE NOTATION

* At all times.
** During GASEOUS RADWASTE SYSTEM operation.
*** 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).
# 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) or (b) or (c) are performed:

a. Initiate the Preplanned Alternate Sampling Program to monitor the appropriate parameter(s).

b. Place moveable air monitors in-line.

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.

TABLE 3.3-12 (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 to monitor the appropriate parameter(s)when it is needed.

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

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE IS REQUIRED
1. GASEOUS RADWASTE SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release RU-12	P	P	R(3)	Q(1),(2),P###	#
b. Flow Rate Monitor	P	N.A.	R	Q,P###	#
2. GASEOUS RADWASTE SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor (continuous)	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (sequential)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor (continuous)	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (sequential)	D	N.A.	Q(5)	M	**

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	*

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
5. FUEL BUILDING VENTILATION SYSTEM (RU-145 and RU-146)					
a. Noble Gas Actvity 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	##

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

* At all times.

** During GASEOUS RADWASTE SYSTEM operation.

*** 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).

\# 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.

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.

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.

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).

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.

REMOTE SHUTDOWN SYSTEM (Continued)

The parameters selected to be monitored ensure that (1) the condition of the reactor is known, (2) conditions in the RCS are known, (3) the steam generators are available for residual heat removal, (4) a source of water is available for makeup to the RCS, and (5) the charging system is available to makeup water to the RCS.

The OPERABILITY of the remote shutdown system insures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control and power circuits and disconnect switches necessary to eliminate effects of the fire and allow operation of instrumentation, control and power circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shutdown the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR 50.

The alternate disconnect methods or power or control circuits ensure that sufficient capability is available to permit shutdown and maintenance of cold shutdown of the facility by relying on additional operator actions at local control stations rather than at the RSP.

3/4.3.3.6 POST-ACCIDENT MONITORING INSTRUMENTATION

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

The containment high range area monitors (RU-148 & RU-149) and the main steamline radiation monitors (RU-139 A&B and RU-140 A&B) are in Table 3.3-6. The high range effluent monitors and samplers (RU-142, RU-144 and RU-146) are in Table 3.3-13. The containment hydrogen monitors are in Specification 3/4.6.5.1. The Post Accident Sampling System (RCS coolant) is in Table 3.3-6.

The Subcooled Margin Monitor (SMM), the Heat Junction Thermocouple (HJTC), and the Core Exit Thermocouples (CET) comprise the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737, the Post TMI-2 Action Plan. The function of the ICC instrumentation is to enhance the ability of the plant operator to diagnose the approach to existance of, and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37. These are not required by the accident analysis, nor to bring the plant to Cold Shutdown.

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.

REACTOR COOLANT SYSTEM

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.

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*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.

REACTOR COOLANT SYSTEM

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.

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 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.

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

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.

---

*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.

#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.

REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

OPERATING

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.

3/4.4.3 PRESSURIZER

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3.1 The pressurizer shall be OPERABLE with a minimum steady-state water level of greater than or equal to 27% indicated level (425 cubic feet) and a maximum steady-state water level of less than or equal to 56% indicated level (948 cubic feet) and at least two groups of pressurizer heaters capable of being powered from Class 1E buses each having a minimum capacity of 125 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of the above required pressurizer heaters OPERABLE, restore at least two groups 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 the pressurizer otherwise inoperable, restore the pressurizer to OPERABLE status within 1 hour, or be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.1.2 The capacity of the above required groups of pressurizer heaters shall be verified to be at least 125 kW at least once per 92 days.

4.4.3.1.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss-of-offsite power:

a. The pressurizer heaters are automatically shed from the emergency power sources, and

b. The pressurizer heaters can be reconnected to their respective buses manually from the control room.

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 verifed locked open at least once per 31 days.

4.4.3.2.3 The auxiliary spray valves shall be cycled at least once per 18 months.

condition of the tubing. This inspection was performed prior to the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.4.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation and shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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.

TABLE 3.4-2

REACTOR COOLANT SYSTEM CHEMISTRY

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm
CHLORIDE	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm
FLUORIDE	$\leq$ 0.10 ppm	$\leq$ 1.00 ppm

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

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once per 72 hours
CHLORIDE	At least once per 72 hours
FLUORIDE	At least once per 72 hours

*Not required with T_{cold} less than or equal to 250°F

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:

MODES 1, 2, and 3*:

a. With the specific activity of the primary coolant greater than 1.0 microcurie/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.

b. 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.

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.

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.

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

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, when ever 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.

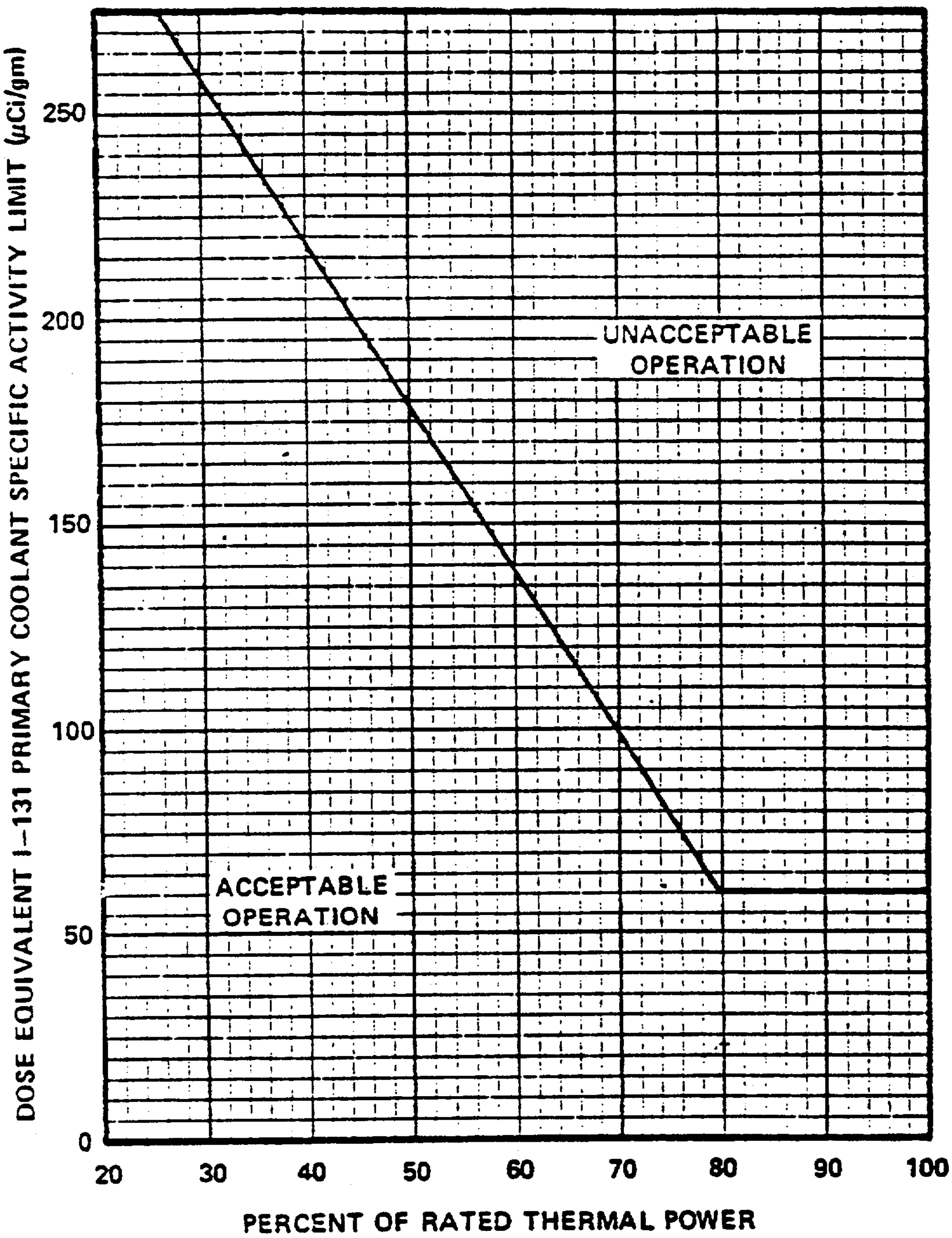

FIGURE 3.4-1

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.

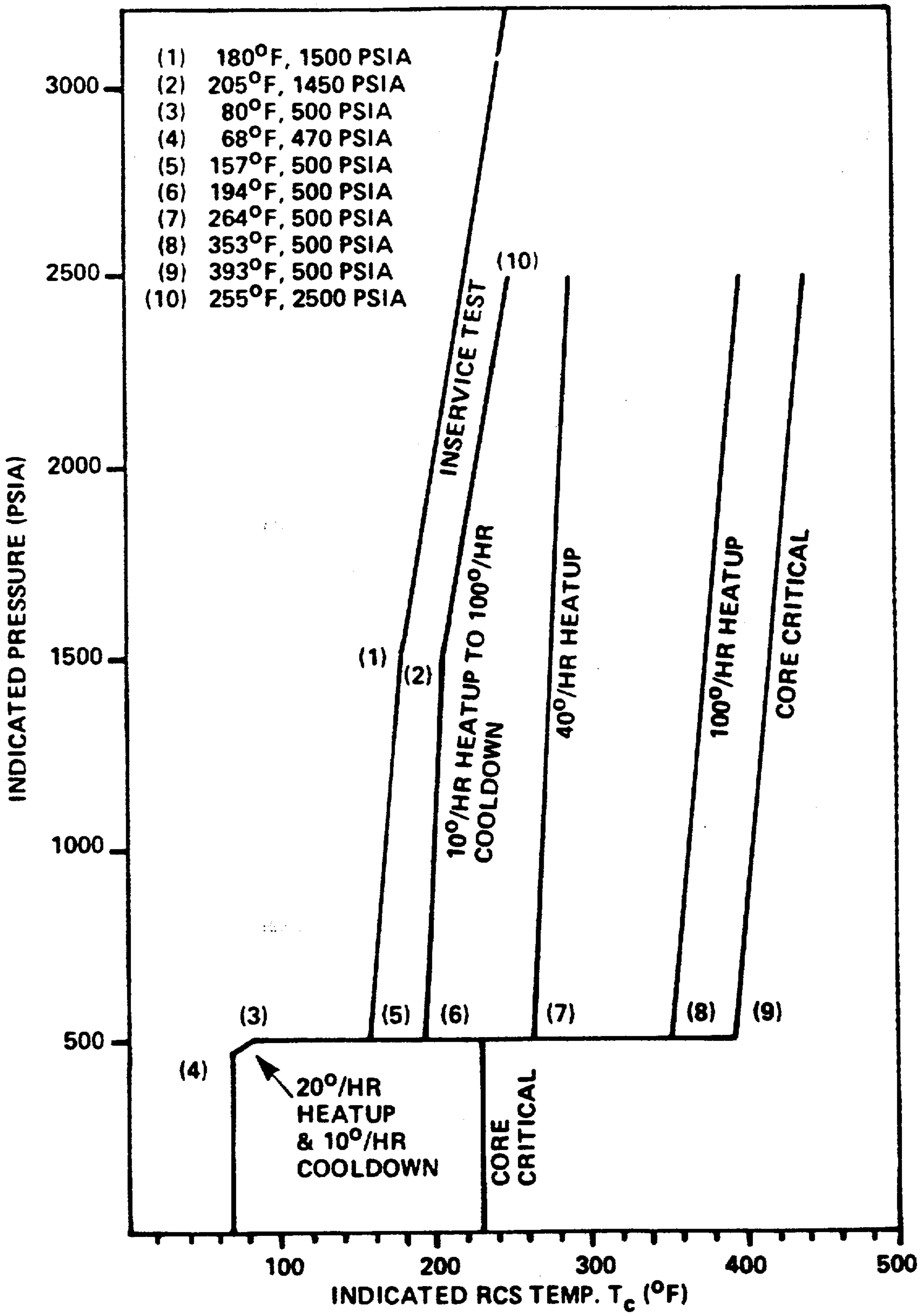

FIGURE 3/4 3.4-2

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

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

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.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

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.

REACTOR COOLANT SYSTEM

3/4.4.9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.9 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.9.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 210°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 0, October 27, 1971.

REACTOR COOLANT SYSTEM

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 Both reactor coolant system vent paths shall be operable and closed at each of the following locations:

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 vent paths OPERABLE, from either location restore both paths at that location 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 reactor coolant system vent paths OPERABLE, from either location restore at least one path at that location 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

4.4.10 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.

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.

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 capsules and the frequencies for removing and testing these capsules are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

(FORGINGS)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a)(b) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	@ 50 ft - lb	MINIMUM UPPER SHELF C_v ENERGY FOR LONGITUDINAL DIRECTION-ft lb
128-101	M-6703-1	SA 508-CL2	Inlet Nozzle	-20	0	+20	+60	N.A.
128-101	M-6703-2	SA 508-CL2	Inlet Nozzle	+10	+10	-25	+10	N.A.
128-101	M-6703-3	SA 508-CL2	Inlet Nozzle	-10	-10	-27	+18	N.A.
128-101	M-6703-4	SA 508-CL2	Inlet Nozzle	0	0	+5	+42	N.A.
131-102	M-4307-1	SA 508-CL2	Outlet Nozzle Safe End	-10	+10	+30	+68	N.A.
131-102	M-4307-2	SA 508-CL2	Outlet Nozzle Safe End	-10	+10	+30	+68	N.A.
128-501	M-6708-1	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-2	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-10	+10	N.A.
128-501	M-6708-3	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-501	M-6708-4	SA 508-CL2	Inlet Nozzle Extension	+20	+20	-20	+20	N.A.
128-301	M-4304-1	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
128-301	M-4304-2	SA 508-CL2	Outlet Nozzle	-10	-10	-35**	-10**	N.A.
131-101	M-6712-1	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-2	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+10	+45	N.A.
131-101	M-6712-3	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
131-101	M-6712-4	SA 508-CL1	Inlet Nozzle Safe End	-10	-10	+7	+50	N.A.
126-101	M-6705-1	SA 508-CL2	Vessel Flange	-70	-70	-78	-28	N.A.
106-101	M-6706-1	SA 508-CL2	Closure Head Flange	-70	-70	-80	-54	N.A.

N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values.
** = Average of three test results.
(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
(b) = 0° and 180° specimens had the same values.

TABLE B 3/4.4-1 (Continued)

REACTOR VESSEL TOUGHNESS

(PLATES)

PIECE NO.	CODE NO.	MATERIAL	VESSEL LOCATION	DROP WEIGHT RESULTS (°F)	RT_{NDT} (a) (°F)	TEMPERATURE OF CHARPY V-NOTCH* @ 30 ft - lb	TEMPERATURE OF CHARPY V-NOTCH* @ 50 ft - lb	MINIMUM UPPER SHELF C_v ENERGY FOR LONGITUDINAL DIRECTION-ft lb
142-102	M-4311-1	SA 533-GRB-CL1	Lower Shell Plate	-10	-10	-6	+40	134
142-102	M-4311-2	SA 533-GRB-CL1	Lower Shell Plate	-40	-40	-24	-8	127
142-102	M-4311-3	SA 533-GRB-CL1	Lower Shell Plate	-20	-20	-7	+14	142
124-102	M-6701-1	SA 533-GRB-CL1	Intermed. Shell Plate	-40	+30	+44	+90	83
124-102	M-6701-2	SA 533-GRB-CL1	Intermed. Shell Plate	-50	+40	+56	+98	96
124-102	M-6701-3	SA 533-GRB-CL1	Intermed. Shell Plate	-30	+40	+39	+89	100
122-102	M-6701-4	SA 533-GRB-CL1	Upper Shell Plate	-30	+60	+82	+120	N.A.
122-102	M-6701-5	SA 533-GRB-CL1	Upper Shell Plate	-30	+40	+49	+98	N.A.
122-102	M-6701-6	SA 533-GRB-CL1	Upper Shell Plate	-30	+40	+42	+96	N.A.
102-102A	M-6709-1	SA 533-GRB-CL1	Closure Head Dome	-20	+10	+36	+66	N.A.
102-102B	M-6709-2	SA 533-GRB-CL1	Closure Head Dome	-70	-20	+4	+37	N.A.
150-102	M-6715-1	SA 533-GRB-CL1	Bottom Head Dome	-30	-30	+2	+30	N.A.
150-102	M-6715-2	SA 533-GRB-CL1	Bottom Head Dome	-40	-10	+26	+50	N.A.

(a) = Determined per applicable ASME-BPV-Code Sect. III, Subsection NB, Article NB-2331-(a-1,2,3).
N.A. = Not Applicable (no minimum upper shelf requirement).
* = Lower bound curve values of transverse specimens.

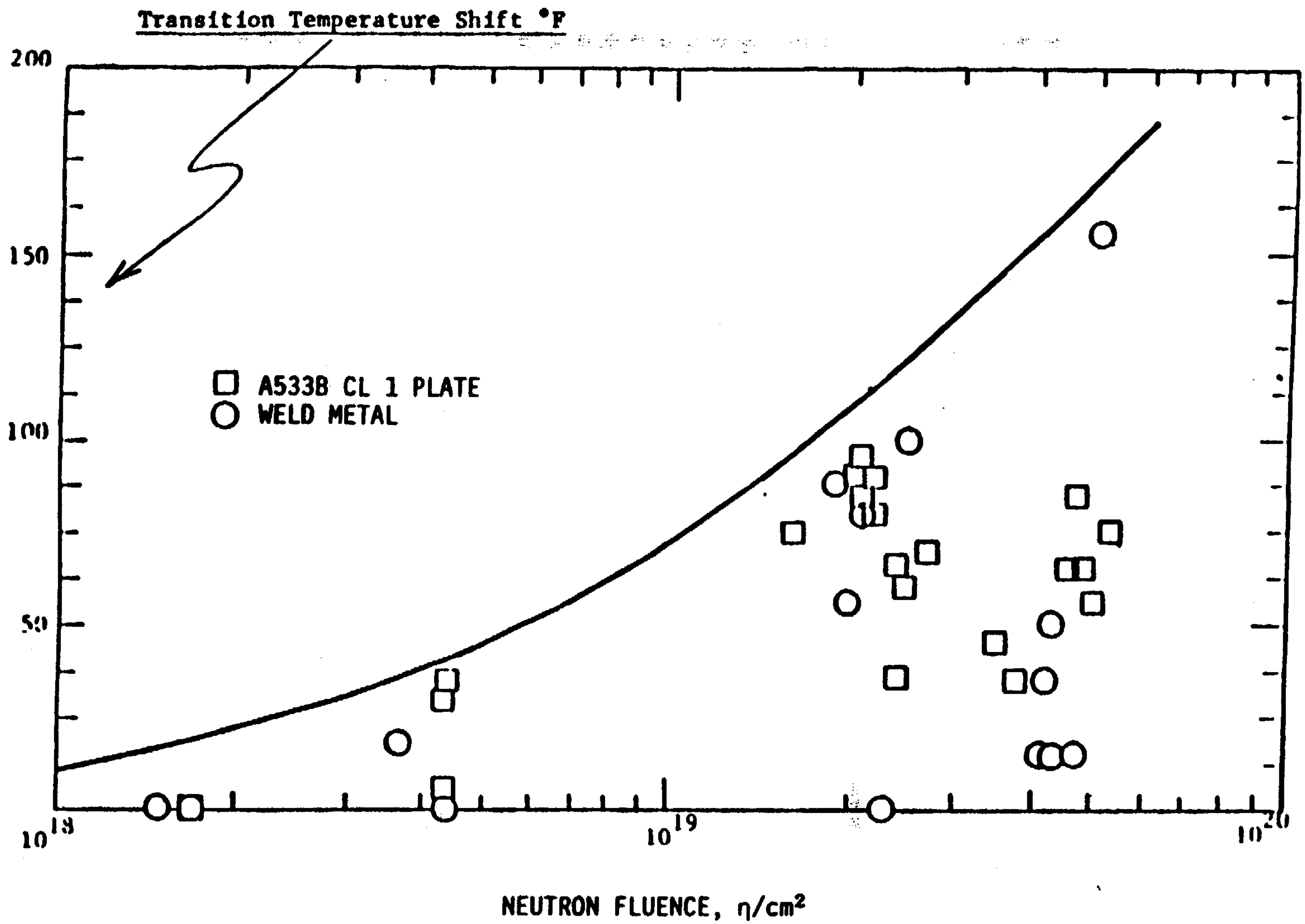

FIGURE B 3/4.4-1

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. 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.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737.

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.

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 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.

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

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.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that each safety injection tank isolation valve is open and the nitrogen vent valves are closed.

b. At least once per 31 days and within 6 hours after each solution level increase of greater than or equal to 7% of tank narrow range level by verifying the boron concentration of the safety injection tank solution is between 2000 and 4400 ppm.

c. At least once per 31 days when the RCS pressure is above 700 psig, by verifying that power to the isolation valve operator is removed.

d. At least once per 18 months by verifying that each safety injection tank isolation valve opens automatically under each of the following conditions:

   1. When an actual or simulated RCS pressure signal exceeds 515 psia, and

   2. Upon receipt of a safety injection actuation (SIAS) test signal.

e. At least once per 18 months by verifying OPERABILITY of RCS-SIT differential pressure alarm by simulating RCS pressure > 715 psia with SIT pressure < 600 psig.

f. At least once per 18 months, when SITs are isolated, by verifying the SIT nitrogen vent valves can be opened.

g. At least once per 31 days, by verifying that power is removed from the nitrogen vent valves.

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. 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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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. SIC-HV 321
2. SIB-UV 625, SIB-UV 307	2. SID-HV 331
3. SIA-UV 635	
4. SIA-UV 645	

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 ± 20 gpm

2. Cold Leg, flow equal to 545 ± 20 gpm

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.

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*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.

REFUELING WATER TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 8.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The limit on the RWT solution temperature ensures that the assumptions used in the LOCA analyses remain valid.

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, 49.5 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 4.6.1.7.2 and 4.6.1.7.3.

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.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 49.5 psig.

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

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days, or

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

3. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage to be less than or equal to 0.01 L_a when determined with the volume between the door seals pressurized to greater than or equal to 14.5 $\pm$ 0.5 psig, for at least 15 minutes,

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

SURVEILLANCE REQUIREMENTS (Continued)

b. By conducting overall air lock leakage tests at not less than P_a, 49.5 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

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.3 and 2.5 psig.

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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. Nominal Elevation 85'0"

b. Nominal Elevation 85'0"

c. Nominal Elevation 126'0"

d. Nominal Elevation 126'0"

e. Nominal Elevation 145'0"

f. Nominal Elevation 188 0"

g. Nominal Elevation 188 0"

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.

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

ACTION:

a. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6 except for Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 15 days, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the structural integrity at a level below the acceptance criteria of Specification 4.6.1.6.2a.4), restore the containment vessel to the required level of integrity within 72 hours, perform an engineering evaluation of the containment vessel structural integrity and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2; or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment vessel shall be demonstrated at the end of 1, 3 and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. All of the acceptance testing of tendon and visual examinations of end anchorages, adjacent concrete surfaces and containment vessel surfaces shall be performed sequentially and within the same time frame.

4.6.1.6.2 The structural integrity of the tendons shall be demonstrated by:

a. Determining from a random but representative sample of at least 10 tendons (6 hoop and 4 inverted U) that each group (hoop, and inverted U) has an observed lift-off force within the predicted limits for that group. For each subsequent inspection one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

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 257 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.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on a safety injection actuation (SIAS) and on a containment spray actuation (CSAS) test signal.

e. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

a. A 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.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 5 years by verifying each solution flow rate from the following drain connections in the iodine removal system:

1. SIA-V253 Pump discharge line 0.63 ± 0.02 gpm.

2. SIB-V254 Pump discharge line 0.63 ± 0.02 gpm.

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.

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*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.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.3 The isolation time of each power operated or automatic valve of Sections A, B and C of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.3.4 The check valves specified in Section D of Table 3.6-1 shall be demonstrated OPERABLE pursuant to 10 CFR 50, Appendix J, with the exception of those check valves footnoted as "Not Type C Tested."

4.6.3.5 The isolation valves specified in Sections E, F, and G of Table 3.6-1 shall be demonstrated OPERABLE as required by Specification 4.0.5 and the Surveillance Requirements associated with those Limiting Conditions for Operation pertaining to each valve or system in which it is installed. Valves secured** in their actuated position are considered operable pursuant to this specification.

4.6.3.6 The manual isolation valves specified in Section H of Table 3.6-1 shall be demonstrated OPERABLE pursuant to Surveillance Requirement 4.6.1.1.a of Specification 3.6.1.1.

**Locked, sealed, or otherwise prevented from unintentional operation.

TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

VALVE NUMBER	PENETRATION NUMBER	FUNCTION	MAXIMUM ACTUATION TIME (SECONDS)
		A. CONTAINMENT ISOLATION (CIAS)	
RDA-UV 023	9	Containment radwaste sump pump to LRS holdup tank	30
RDB-UV 024	9	Containment radwaste sump pump to LRS holdup tank	5
RDB-UV 407	9	Containment radwaste sump post-accident sampling system	5
SGB-HV 200#	11	Downcomer feedwater chemical injection	1
SGB-HV 201#	12	Downcomer feedwater chemical injection	1
SIA-UV 708#	23	Containment recirc sump to post-accident sampling system	5
HCB-UV 044	25A	Containment air radioactivity monitor (inlet)	12
HCA-UV 045	25A	Containment air radioactivity monitor (inlet)	12
HCA-UV 046	25B	Containment air radioactivity monitor (outlet)	12
HCB-UV 047	25B	Containment air radioactivity monitor (outlet)	12
GAA-UV 002	29	N_2 to steam generator and reactor drain tank	10
GAA-UV 001	30	N_2 to SI tanks	10

#Not Type C Tested

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	Pressurizer liquid sample line	5
SSA-UV 204	42A	Pressurizer liquid sample line	5
SSB-UV 202	42B	Pressurizer steam space sample line	5
SSA-UV 205	42B	Pressurizer steam space sample line	5
SSB-UV 200	42C	Hot leg sample line	5
SSA-UV 203	42C	Hot leg sample line	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 tank drain	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.

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.

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.4.2 Two portable independent containment hydrogen recombiner systems shared among the three units shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or meet the requirements of Specification 3.6.4.3, or be in at least HOT STANDBY within the next 6 hours.*

SURVEILLANCE REQUIREMENTS

4.6.4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by:

1. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure and control console.

2. Operating the recombiner to include the air blast heat exchanger fan motor and enclosed blower motor continuously for at least 30 minutes at a temperature of approximately 800°F reaction chamber temperature.

b. At least once per year by performing a CHANNEL CALIBRATION of recombiner instrumentation to include a functional test of the recombiner at 1200°F (± 50°F) for at least four hours.

*Prior to March 30, 1986 or until the completion of the environmental qualification modifications to the hydrogen recombiner system, whichever occurs first, the provisions of Specification 3.0.4 are not applicable during implementation of the environmental qualification modifications to the hydrogen recombiner system when the containment hydrogen purge cleanup system described in Specification 3.6.4.3 is OPERABLE.

CONTAINMENT SYSTEMS

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.**

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

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

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 49.5 psig. The limit of 2.5 psig for initial positive containment pressure will limit the total pressure to 49.5 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 49.5 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 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 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, main steam atmospheric dump valves, and main steam isolation valves. The main steam safety valves have very high pressure setpoints to actuate and are covered by Specification 3/4.7.1.1. The atmospheric dump valves and the main steam isolation valves are covered by Specifications 3/4.7.1.6 and 3/4.7.1.5, 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.

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).

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 42-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 42-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The containment spray system and the containment cooling system are redundant to each other in providing post-accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure adequate chemical available to remove iodine from the containment atmosphere following a LOCA.

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 may proceed provided that within 4 hours, either all the inoperable valves are restored to OPERABLE status or the 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).

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOPS

VALVE NUMBER	S/G No. 1	S/G No. 2	LIFT SETTING (±1%)*	MINIMUM RATED CAPACITY**
a.	SGE PSV 572	SGE PSV 554	1250 psig	941,543 lb/hr
b.	SGE PSV 579	SGE PSV 561	1250 psig	941,543 lb/hr
c.	SGE PSV 573	SGE PSV 555	1290 psig	971,332 lb/hr
d.	SGE PSV 578	SGE PSV 560	1290 psig	971,332 lb/hr
e.	SGE PSV 574	SGE PSV 556	1315 psig	989,950 lb/hr
f.	SGE PSV 575	SGE PSV 557	1315 psig	989,950 lb/hr
g.	SGE PSV 576	SGE PSV 558	1315 psig	989,950 lb/hr
h.	SGE PSV 577	SGE PSV 559	1315 psig	989,950 lb/hr
i.	SGE PSV 691	SGE PSV 694	1315 psig	989,950 lb/hr
j.	SGE PSV 692	SGE PSV 695	1315 psig	989,950 lb/hr

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

**Capacity is rated at lift setting +3% accumulation.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months during shutdown by:

  1. Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an auxiliary feedwater actuation test signal.

  2. Verifying that each pump that starts automatically upon receipt of an auxiliary feedwater actuation test signal will start automatically upon receipt of an auxiliary feedwater actuation test signal.

c. Prior to startup following any refueling shutdown or cold shutdown of 30 days or longer, by verifying on a STAGGERED TEST BASIS (by means of a flow test) that the normal flow path from the condensate storage tank to each of the steam generators through one of the essential auxiliary feedwater pumps delivers at least 750 gpm at 1270 psia or equivalent.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or MODE 4 for the turbine-driven pump.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with an indicated level of at least 25 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

b. Demonstrate the OPERABILITY of the reactor makeup water tank as a backup supply to the essential 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 essential auxiliary feedwater pumps by verifying:

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).

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

#Not applicable when cooldown is in progress.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

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

ACTION:

MODE 1:

With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least MODE 2 within the next 6 hours.

MODES 2, 3, and 4:

With one main steam line isolation valve inoperable, subsequent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5.1 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 4.6 seconds when tested pursuant to Specification 4.0.5.

4.7.1.5.2 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 4.7.1.5.1 provided the testing is performed within 12 hours after achieving normal operating steam pressure and normal operating temperature for the secondary side to perform the test.

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 $\geq$ 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.

PLANT SYSTEMS

3/4.7.6 ESSENTIAL CHILLED WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 At least two independent essential chilled water loops shall be OPERABLE.

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

ACTION:

a. With only one essential chilled water loop OPERABLE, restore at least two loops 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.

b. With only one essential chilled water system OPERABLE:

   1. Within 1 hour verify that the normal HVAC system is providing space cooling to the vital power distribution rooms that depend on the inoperable essential chilled water system for space cooling, and

   2. Within 8 hours establish OPERABILITY of the safe shutdown systems which do not depend on the inoperable essential chilled water system (one train each of boration, pressurizer heaters and auxiliary feedwater), and

   3. Within 24 hours establish OPERABILITY of all required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE essential chilled water system for space cooling.

   If these conditions are not satisfied within the specified time, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1 At least two essential chilled water loops shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

4.7.6.2 Once per 18 months during shutdown, verify that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is locked, sealed, or otherwise secured in position, is in its correct position.

PLANT SYSTEMS

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 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:

ELECTRICAL POWER SYSTEMS

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

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

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

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 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:

SURVEILLANCE REQUIREMENTS (Continued)

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. 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.

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. A visual inspection of the systems shall be made within 6 months following such an event. In addition to satisfying

*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.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one D.C. train as listed in Table 3.8-1 shall be OPERABLE and energized.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With a required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required D.C. train to OPERABLE status as soon as possible.

b. With a required charger inoperable, either provide charging capability to the affected channel with the associated backup battery charger, or demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

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-M1533	E-NHN-M1502B	REACTOR CAVITY FAN D DISCH DAMPER M-HCN-M02D
E-NHN-M1534	E-NHN-M1535	CTMT BLDG MONO HOIST 1 TON M-ZCN-009
E-NHN-M1517	E-NHN-M1535	REACTOR COOLANT OIL LIFT PUMP M-RCN-P02A
E-NHN-M1902	E-NHN-M1917A	REACTOR CAVITY NORM CLG FAN M-HCN-A03A
E-NHN-M1904	E-NHN-M1917B	REACTOR CAVITY NORM CLG FAN M-HCN-A03C
E-NHN-M1907	E-NHN-M1917	CEDM NORM ACU-A HEXCH OUTLET VLV J-NCN-HV-485
E-NHN-M1911	E-NHN-M1917	CTMT NORM ACU-C CHILLED WTR INLET VLV J-WCN-HV-59
E-NHN-M1912	E-NHN-M1917	CTMT NORM ACU-A CHILLED WTR INLET VLV J-WCN-HV-57
E-NHN-M2008	E-NHN-M2010	CEDM NORM ACU-B HEXCH OUTLET VLV J-NCN-HV-486
E-NHN-M2003	E-NHN-M2010	CTMT NORM ACU-B CHILL WATER INLET VLV J-WCN-HV-58
E-NHN-M2004	E-NHN-M2010	CTMT NORM ACU-D CHILL WATER INLET VLV J-WCN-HV-60
E-NHN-M2006	E-NHN-M2010A	REACTOR CAVITY NORM CLG FAN M-HCN-A03B
E-NHN-M2007	E-NHN-M2016	REACTOR CAVITY NORM CLG FAN M-HCN-A03D
E-NHN-M2803	E-NHN-M2827A	CEDM ACU C INTAKE DAMPER M-HCN-M03C
E-NHN-M2804	E-NHN-M2827A	CEDM ACU D INTAKE DAMPER M-HCN-M03D

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 COOLANT 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-G04A, 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-G04A, 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-NHN-M7102	E-NHN-M7104	CONTAINMENT NORMAL ACU A DISCHARGE DAMPER M-HCN-M01A
E-NHN-M7103	E-NHN-M7104	CONTAINMENT NORMAL ACU C DISCHARGE DAMPER M-HCN-M01C
E-NHN-M7114	E-NHN-7113	PZR NORMAL COOLING FAN M-HCN-A06A
E-NHN-M2816	E-NHN-M2832C	CTMT BLDG MONORAIL HOIST-2 TON M-ZCN-G08
E-NHN-M2834A	E-NHN-M2832C	MOVABLE INCORE DETECTOR DRIVE MACH #2 M-RIN-M03B
E-NHN-M7202	E-NHN-M7204	CTM NORM ACU B DISCH DAMPER M-HCN-M01B
E-NHN-M7203	E-NHN-M7204	CTM NORM ACU D DISCH DAMPER M-HCN-M01D
E-NHN-M7214	E-NHN-M7213	PZR NORMAL COOLING FAN M-HCN-A06B
E-PGA-L31E2	E-NGN-B31E2 (FUSE)	CONTAINMENT NORMAL ACU FAN M-HCN-A01A
E-PGA-L31E3	E-NGN-B31E3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02A
E-PGB-L32E3	E-NGN-B32E3 (FUSE)	PRESSURIZER BACKUP HEATERS M-RCE-B18, B10, A5
E-PGB-L32E2	E-NGN-B32E2 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02B
E-PGA-L33D2	E-NGN-B33D2 (FUSE)	CONTAINMENT NORMAL ACU FAN M-HCN-A01C
E-PGA-L33D4	E-NGN-B33D4 (FUSE)	PRESSURIZER BACKUP HTR, M-RCE B1, B9, A14
E-PGA-L33D3	E-NGN-B33D3 (FUSE)	CEDM NORMAL ACU FAN M-HCN-A02C

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-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

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 J-GRA UV1
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-D0102	E-NGN-L11C2	PZR BU HTR M-RCE-B07, B13, A01
E-NHN-D2614	E-NHN-M2618	TO OPERATION CAMERA JB# 1
E-RCN-D0101	E-NGN-L11C2	PZR BU HTR M-RCE-B03, A09, A15
E-RCN-D0301	E-NGN-L11C3	PZR BU HTR M-RCE-B04, A11, A16
E-RCN-D0302	E-NGN-L11C3	PZR BU HTR M-RCE-A02, A07, A13
E-RCN-D0201	E-NGN-L12C2	PZR BU HTR M-RCE-B06, B12, A18
E-RCN-D0202	E-NGN-L12C2	PZR BU HTR M-RCE-B16, A04, A08
E-RCN-D0401	E-NGN-L12C3	PZR BU HTR M-RCE-B15, A03, A10
E-RCN-D0402	E-NGN-L12C3	PZR BU HTR M-RCE-A17, A06, A12

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 (FUSE)	E-NGN-L11C4	PROPORTIONAL HTR BANK M-RCE-B2, B8, B14
J-RCN-PC100B (FUSE)	E-NGN-L12C4	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
CEA 12 CB104	F110, F111, F112	CEA 12
CEA 07 CB101	F101, F102, F103	CEA 07
CEA 09 CB102	F104, F105, F106	CEA 09
CEA 11 CB103	F107, F108, F109	CEA 11
CEA 13 CB104	F110, F111, F112	CEA 13
CEA 74 CB101	F101, F102, F103	CEA 74
CEA 76 CB102	F104, F105, F106	CEA 76
CEA 78 CB103	F107, F108, F109	CEA 78
CEA 80 CB104	F110, F111, F112	CEA 80
CEA 75 CB101	F101, F102, F103	CEA 75
CEA 77 CB102	F104, F105, F106	CEA 77
CEA 79 CB103	F107, F108, F109	CEA 79
CEA 81 CB104	F110, F111, F112	CEA 81
CEA 22 CB101	F101, F102, F103	CEA 22
CEA 24 CB102	F104, F105, F106	CEA 24
CEA 26 CB103	F107, F108, F109	CEA 26
CEA 28 CB104	F110, F111, F112	CEA 28
CEA 23 CB101	F101, F102, F103	CEA 23
CEA 25 CB102	F104, F105, F106	CEA 25

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 M-RCE-P01BH
E-NHN-D15-02	E-NHN-M1526	REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01DH
E-NHN-D15-06	E-NHN-M1526	CONTAINMENT PREACCESS NORMAL AFU FAN MOTOR SPACE HEATER M-HCN-F01BH
E-NHN-D10-01	E-NHN-M1027	REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01AH
E-NHN-D10-02	E-NHN-M1027	REACTOR COOLANT PUMP MOTOR SPACE HEATER M-RCE-P01CH
E-NHN-D10-20	E-NHN-M1027	STEAM GENERATOR WET LAYUP PUMP MOTOR SPACE HEATER M-SGN-P01BH
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

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-A01DH
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-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
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

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-ZJB-C03 (FUSE)	E-PKB-D2211	SI TANK CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-638
E-ZJB-C03 (FUSE)	E-PKB-D2211	HOT LEG INJ CHECK VLV LEAKAGE LINE ISO VLV J-SIB-UV-332
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	REACTOR COOLANT VENT VALVE J-RCA-HV-101
E-ZAA-C03 (FUSE)	E-PKA-D2109	REGENERATIVE HEAT EXCH TO AUX SPRAY VALVE J-CHA-HV-205
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
E-ZAA-C04 (FUSE)	E-PKA-D2130	CTMT PRG PWR ACCESS MODE ISO VLV J-CPA-UV-4A

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
E-ZAA-C05 (FUSE)	E-PKA-D2114	STEAM GEN BLOWDOWN CTMT ISOLATION VALVE J-SGA-UV-500P
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-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

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
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-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
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-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-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

TABLE 3.8-2 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PRIMARY DEVICE NUMBER	BACKUP DEVICE NUMBER	SERVICE DESCRIPTION
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
E-ZAN-C02 (FUSE)	E-NKN-D4216	SEAL INJECT VALVES TO RCP J-CHE-FV-243
E-ZAN-C02 (FUSE)	E-NKN-D4216	REGEN HEAT EXCH TO CHARGING LINE VALVE J-CHE-PDV-240
E-PGB-L32E2 (FUSE)	E-PGB-L32E2 (FUSE)	CEDM NORM ACU FAN - B M-HCN-A02B
E-PGB-L34D2 (FUSE)	E-PGB-L34D2 (FUSE)	CTMT NORM ACU FAN - D M-HCN-A01D
E-PKC-M4322	E-PKC-M4304	SAFETY INJECTION SHUTDOWN COOLING ISOLATION VALVE J-SIC-UV-653
E-PKD-M4422-1	E-PKC-M4404	SAFETY INJECTION SHUTDOWN COOLING ISOLATION VALVE J-SIC-UV-654
E-PNA-D2519 (FUSE)	E-PNA-D25	MAIN PANEL BREAKER SHUTDOWN COOLING ISOLATION VALVE J-SIB-UV-651 - INDICATION LIGHTS
E-PNB-D2619 (FUSE)	E-PNB-D26	MAIN PANEL BREAKER SHUTDOWN COOLING ISOLATION VALVE J-SIB-UV-652 - INDICATION LIGHTS
E-NHN-D1506	E-NHN-M1526	CTMT PRE-ACCESS NORMAL AFU FAN MOTOR HEATER M-HCN-F01BH

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION AND BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve shown in Table 3.8-3 shall be bypassed continuously or under accident conditions, as applicable, by an OPERABLE device integral with the motor starter.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device, take administrative action to continuously bypass the thermal overload within 8 hours or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2.1 The thermal overload protection for the above required valves shall be verified to be bypassed continuously or under accident conditions, as applicable, by an OPERABLE integral bypass device by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overloads which are normally in force during plant operation and bypassed under accident conditions and by verifying that the thermal overload protection is bypassed for those thermal overloads which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing:

a. At least once per 18 months, and

b. Following maintenance on the motor starter.

4.8.4.2.2 The thermal overload protection for the above required valves which are continuously bypassed shall be verified to be bypassed following testing during which the thermal overload protection was temporarily placed in force.

TABLE 3.8-3

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (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 (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.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (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 (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.

TABLE 3.8-3 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD

PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	BYPASS DEVICE (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 (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 (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

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 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.

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 float on 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 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2150 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 26 gpm of a solution containing $\geq$ 4000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2150 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

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*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

3/4.9.2 INTRUMENTATION

LIMITED CONDITION FOR OPERATION

3.9.2 As a minimum, two startup channel 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

4.9.2 Each startup channel 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 FUNTIONAL TEST at least once per 7 days.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2000 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

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 OPERABLE 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.

LOW WATER LEVEL

LIMITED 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.

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge valve isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the containment purge valve isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment purge valve isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge valve isolation occurs on manual initiation and on CPIAS.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITED 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.

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.

3/4.9.12 FUEL BUILDING ESSENTIAL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12* Two independent fuel building essential ventilation systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel building essential ventilation system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel building essential ventilation system is capable of being powered from an OPERABLE emergency power source. Restore the inoperable fuel building essential ventilation system to OPERABLE status within 7 days or suspend all operations involving movement of fuel within the storage pool or operation of the fuel handling machine over the storage pool.

b. With no fuel building essential ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel building essential ventilation system is restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel building essential ventilation systems 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:

*CAUTION - Reference Specification 3.7.8 page 3/4 7-19

3/4.9 REFUELING OPERATIONS

BASES

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

The OPERABILITY of the startup channel 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.

/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: 1) the machine will be used for movement of fuel assemblies, (2) the machine as sufficient load capacity to lift a fuel assembly, and (3) the core internals .nd pressure vessel are protected from excessive lifting force in the event .hey are inadvertently engaged during lifting operations.

:/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a 'uel assembly, CEA and associated handling tool over other fuel assemblies in :he storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation, and circulating reactor coolant at a flow rate equal to or greater than 4000 gpm ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during the REFUELING MODE, (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification, and (3) the ΔT across the core will be maintained at less than 75°F during the REFUELING MODE. The required flowrate of $\geq$ 4000 gpm ensures that 240 hours after reactor shutdown sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 135°F as required during REFUELING MODE; this assumes a shutdown cooling heat exchanger cooling water flowrate of 14000 gpm, a cooling water inlet temperature of $\leq$ 105°F at $\geq$ 27 1/2 hours after reactor shutdown, and the decay heat curve of CESSAR-F Figure 6.2.1-1 and reactor operation for two years at 4000 MWt.

tion

Without a shutdown cooling train in operation steam may be generated; therefore, the containment should be sealed off to prevent escape of any radioactivity, and any operations that would cause an increase in decay heat should be secured.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange, ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), or the reactor is subcritical by at least the reactivity equivalent of the highest CEA worth.

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

ACTION:

a. With any full-length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 26 gpm of a solution containing greater than or equal to 4000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length and part-length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 When in MODE 3 or MODE 4, the reactor shall be determined to be subcritical by at least the reactivity equivalent of the highest estimated CEA worth or the reactivity equivalent of the highest estimated CEA worth is available for trip insertion from OPERABLE CEAs at least once per 2 hours by consideration of at least the following factors:

a. Reactor Coolant System boron concentration,

b. CEA position,

c. Reactor Coolant System average temperature,

d. Fuel burnup based on gross thermal energy generation,

e. Xenon concentration, and

f. Samarium concentration.

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*Operation in MODE 3 and MODE 4 shall be limited to 6 consecutive hours.

#Limited to low power PHYSICS TESTING at the 320°F plateau.

SPECIAL TEST EXCEPTIONS

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, 3.1.3.7, 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, 3.1.3.7, 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, 3.1.2.5, 3.1.3.6, 3.1.3.7, 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 4.2.1.2 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, 3.1.3.7, 3.2.2, 3.2.3, 3.2.7, or the Minimum Channels OPERABLE requirements of I.C.1 (CEA Calculators) of Table 3.3-1 are suspended.

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 radoinuclides 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 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, HSAL-300 (revised annually).
(2) Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemisty" Anal. Chem 40, 586-93 (1968).
(3) Hartwell, J. K., "Detection Limits for Radioisotopic Counting Techniques," Atlatic Richfield Hanford Company Report (ARH-2537 (June 22, 1972).

TABLE 4.11-2 (Continued)

TABLE NOTATION

[b]Analyses shall also be performed following SHUTDOWN, STARTUP, or a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period if 1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and 2) the noble gas activity monitor on the plant vent shows that effluent activity has increased by more than a factor of 3. If the associated noble gas vent monitor is inoperable, samples must be obtained as soon as possible. Analyses shall be performed within a four-hour period. This requirement does not apply to the Fuel Building Exhaust.

[c]Sampling and analyses shall also be performed at least once per 31 days when purging time exceeds 30 days continuous.

[d]Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

[e]Tritium grab samples shall be taken at least monthly from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

[f]The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

[g]The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported in the Semiannual Radioactive Effluent Release Report.

TABLE 3.12-1

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]
Airborne			
Radioiodine and particulates	Samples from 5 locations: 3 samples at or near the SITE BOUNDARIES (#14A, 15, 21) in different sectors of the highest calculated annual average ground level D/Q.* 1 sample (#40) from areas of special interest, which is from the vicinity of a community having the highest calculated annual average D/Q. 1 sample (#6) from a control location 15-30 km (10-20 mi) distant and in the least prevalent wind direction.[f]	Continuous sampling collected weekly,[c] or more frequently if required by dust loading	Gross beta weekly;[d] I-131 weekly; gamma isotopic analysis of composite (by location) quarterly
Direct radiation[b]	40 stations (#6-45) with two or more dosimeters for measuring dose rate continuously, placed as follows: an inner ring of stations at the site boundary and an outer ring in the 4-to-5 mi range from the site with a station in each sector of each ring, except the WNW sector, which is inaccessible (16 sectors x 2 rings minus 1 = 31 stations). 7 additional stations are in local schools and population centers; 2 other stations are used as controls.	Quarterly	Gamma dose quarterly

*D/Q refers to average annual relative ground deposition rate.

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 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

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 (#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 (#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 3.12-1 (Continued)

TABLE NOTATIONS

[a]The number, media, frequency, and location of sampling may vary from site to site. It is recognized that, at times, it may not be possible or practical to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and submitted for acceptance. Actual locations (distance and direction) from the site shall be provided in Table 7-1 and Figure 7-1 in the ODCM. Refer to Regulatory Guide 4.1, "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants."

[b]Regulatory Guide 4.13 provides guidance for thermoluminescence dosimetry (TLD) systems used for environmental monitoring. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter may be considered to be one phosphor, and two or more phosphors in a packet may be considered as two or more dosimeters. Film badges should not be used for measuring direct radiation.

[c]Canisters for the collection of radioiodine in air are subject to channeling. These devices should be carefully checked before operation in the field or several should be mounted in series to prevent loss of iodine.

[d]Particulate sample filters shall be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air or water is greater than 10 times the yearly mean of control samples for any medium, gamma isotopic analysis should be performed on the individual samples.

[e]Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

[f]The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

[g]Groundwater samples should be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.

[h]The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]Guidance for detection capabilities for thermoluminescent dosimeters used for environmental measurements is given in Regulatory Guide 4.13.

[b]Table 4.12-1 indicates acceptable detection capabilities for radioactive materials in environmental samples. These detection capabilities are tabulated in terms of the lower limits of detection (LLDs). The LLD is defined, for purposes of this guide, as the smallest concentration of radioactive material in a sample that will yield a net count (above system background) that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the "a priori" lower limit of detection as defined above (as picocuries per unit mass or volume).

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 disintegration)

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

2.22 is the number of disintegrations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt for environmental samples is the elapsed time between sample collection (or end of the sample collection period) and time of counting

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.

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

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 resotre 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 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.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers located on site. Each shall have a Bachelor's Degree in engineering or related science and at least two years professional level experience in his field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly to reduce human errors as much as practical, and to detect potential nuclear safety hazards.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Director, Nuclear Safety and Licensing, Plant Manager, and the Manager, Nuclear Safety Group (NSG).

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Director, Nuclear Safety and Licensing.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor (STA) shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The STA shall be onsite and shall be available in the control room within 10 minutes whenever one or more units are in MODE 1, 2, 3, or 4.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANS 3.1-1978 and Regulatory Guide 1.8, September 1975, except for the Radiation Protection and Chemistry Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A training program for the unit staff shall be maintained under the direction of the Director, Site Services or his designee and shall meet or exceed the requirements and recommendations of Section 5.0 of ANS 3.1-1978 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The Plant Review Board shall function to advise the Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PRB shall be composed of the following personnel:

Member:	Engineering Evaluations Manager
Member:	Operations Standards Supervisor
Member:	Mechanical Maintenance Standards Supervisor
Member:	Electrical Maintenance Standards Supervisor
Member:	Operations Managers for Unit 1, Unit 2, Unit 3
Member:	STA Supervisor
Member:	I&C Standards Supervisor
Member:	Radiation Protection and Chemistry Manager
Member:	Quality Systems/Engineering Manager

The Vice President-Nuclear Production shall designate the Chairman and Vice-Chairmen in writing. The Chairman and Vice-Chairmen may be from outside the members listed above provided that they meet ANSI Standard 3.1, 1978.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRB activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRB shall meet at least once per calendar month and as convened by the PRB Chairman, Vice-Chairmen, or his designated alternate.

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.

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 Vice President-Nuclear Production.

g. Review and documentation 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, Plant Manager and NSG of disagreement between the PRB and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

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 Plant Manager, Vice President-Nuclear Production and NSG.

REVIEW (Continued

b. Proposed changes to procedures, equipment, systems or facilities within the power block which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety;

g. All REPORTABLE EVENTS requiring 24 hours written notification;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB.

AUDITS

6.5.3.5 Audits of unit activities shall be performed under the cognizance of the NSG. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training, and qualifications of the unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the NSG or the Vice President-Nuclear Production.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

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.

AUTHORITY

6.5.3.6 The NSG shall report to and advise the Director, Nuclear Safety and Licensing 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 Director, Nuclear Safety and Licensing who will distribute it to the Vice President-Nuclear Production, 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:

a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50, and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PRB, and the results of this review shall be submitted to the Manager 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, Plant Manager and Manager 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 Manager of the NSG and the Vice President-Nuclear Production within 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) 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)

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.

NOTE: The licensee shall perform a secondary water chemistry monitoring and control program that is in conformance with the program discussed in Section 10.3.4.1 of the CESSAR FSAR or another NRC approved program.

m. Post-Accident Sampling System implementation.*

n. Settlement Monitoring Program implementation.

NOTE: The licensee shall maintain a settlement monitoring program throughout the life of the plant in accordance with the program presented in Table 2.5-18 of the PVNGS FSAR or another NRC approved program.

o. CEA Reactivity Integrity Program implementation

NOTE: The licensee shall perform, after initial fuel load or after each reload, either a CEA symmetry test or worth measurements of all full-length CEA groups to address Section 4.2.2 of the PVNGS SER dated November 11, 1981.

p. Fuel Assembly Surveillance Program Implementation

NOTE: The licensee shall perform a fuel assembly surveillance program in conformance with the program discussed in Section 4.2.4 of the PVNGS SER dated November 11, 1981.

6.8.2 Each program or procedure of Specification 6.8.1, and changes thereto, shall be reviewed as specified in Specification 6.5 and approved prior to implementation. Programs, administrative control procedures and implementing procedures shall be approved by the Vice President-Nuclear Production, or designated alternate who is at supervisory level or above. Programs and procedures of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

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

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

b. The change is approved by two members of the plant supervisory staff, at least one of whom is a Shift Supervisor or Assistant Shift Supervisor with an SRO on the affected unit.

c. The change is documented, reviewed in accordance with Specification 6.5.2 and approved by the Director, Standards and Technical Support or cognizant department head, as designated by the Vice President-Nuclear Production, within 14 days of implementation.

*Not required until prior to exceeding 5% of RATED THERMAL POWER.

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the NSG at least once per 24 months:

a. Primary Coolant Sources Outside Containment

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

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

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

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

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

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

PROCEDURES AND PROGRAMS (Continued)

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(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 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

REPORTING REQUIREMENTS (Continued)

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.

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 the 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

*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.

ANNUAL REPORTS (Continued)

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.

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.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, with operational controls as appropriate, and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps**

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

**One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3; discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.8 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Release of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

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.

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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; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

**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.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

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),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM and to the OFFSITE DOSE CALCULATION MANUAL, as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with 10 CFR 50.73.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures of Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the FSAR.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released beyond the SITE BOUNDARY.

e. Records of transient or operational cycles for those unit components identified in Tables 5.7-1 and 5.7-2.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

RECORD RETENTION (Continued)

i. Records of quality assurance activities required by the QA Manual not listed in Section 6.10.1.

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

k. Records of PRB meetings and of NSG activities.

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.9 including the date at which the service life commences and associated installation and maintenance records.

m. Records of audits performed under the requirements of Specifications 6.5.3.5 and 6.8.4.

n. Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

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

6.12 HIGH RADIATION AREA

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

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

HIGH RADIATION AREA (Continued)

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

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

c. A radiation protection qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Radiation Protection Supervisor or his designated alternate in the REP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose greater than 1000 mrem shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or radiation protection supervision. Doors shall remain locked except during periods of access by personnel under an approved REP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose in excess of 1000 mrems*, that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the REP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

6.13 PROCESS CONTROL PROGRAM (PCP)

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

6.13.2 Licensee-initiated changes to the PCP:

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:

*Measurement made at 18 inches from source of radioactivity.

PROCESS CONTROL PROGRAM (PCP) (Continued)

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

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

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee-initiated changes to the ODCM:

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

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s); and

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the radioactive waste systems (liquid, gaseous, and solid):

Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

*Licensees may chose to submit the information called for in this specification as part of the annual FSAR update.

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

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. NPF 41

ARIZONA PUBLIC SERVICE COMPANY, ET AL.

PALO VERDE NUCLEAR GENERATING STATION, UNIT NO. 1

DOCKET NO. STN 50-528

1.0 INTRODUCTION

By letter dated May 25, 1987, as amended by letter dated August 7, 1987, the Arizona Public Service Company (APS) on behalf of itself, the Salt River Project Agricultural Improvement and Power District, Southern California Edison Company, El Paso Electric Company, Public Service Company of New Mexico, Los Angeles Department of Water and Power, and Southern California Public Power Authority (licensees), requested changes to the Technical Specifications (Appendix A to Facility Operating License NPF-41) for the Palo Verde Nuclear Generating Station, Unit 1.

The proposed changes would revise a number of Technical Specifications and can be categorized as those which (1) respond to changes in the regulations or regulatory guidance, (2) are more restrictive than the current Technical Specifications, or (3) are administrative changes since they are either editorial, provide clarification, remove redundancy or correct errors. All of the changes would make those areas of the Technical Specifications consistent with the Technical Specifications previously reviewed and approved by the staff for Palo Verde, Units 2 and 3 (Facility Operating Licenses NPF-51 and NPF-74, respectively).

2.0 DISCUSSION

The proposed amendment consists of approximately 200 changes that are specifically identified by item numbers in the licensees' submital and which can be grouped into the above three categories. A discussion of these changes is presented below:

(a) The proposed changes which respond to changes in the regulations or regulatory gudiance are as follows:

(i) Item 103 deals with Specification 3/4.4.7 regarding specific activity limits for the primary coolant. The proposed change would bring the Action Statement for specific activity in the primary coolant into conformance with Generic Letter 85-19. Item 209 deals with Specification 6.8.1.5 regarding annual reports for the facility. This proposed change would add to the annual reporting requirements the results of primary coolant specific activity analysis in which the primary coolant exceeds the limits of Specification 3/4.4.7. Included with the above changes are the associated administrative changes due to the revised bases section, revised table of contents and renumbered pages; these are Items 6, 7, 8, 16, 17, 18, 19, 21, 22, 104, 105, 106, 108, 110, 112, 113, 114, and 119.

(ii) Item 206 deals with Specification 6.7.1.c regarding the Safety Limit Violation Report. The proposed change would revise the time for submitting such a report from within 14 days to within 30 days, which is in conformance with the requirements specified in 10 CFR 50.73(a).

(b) The proposed changes which are more restrictive than the current Technical Specifications are as follows.

(i) Item 35 deals with the bases section for boration systems (Specification 3/4.1.2). The proposed change would add to the list of components required to perform boron injection by including the volume control tank and an associated valve.

(ii) Item 96 deals with Specification 4.4.3.2 regarding the surveillance requirements for the auxiliary spray system. The proposed change would include surveillance requirements for two additional valves associated with the system.

(iii) Item 126 deals with Specification 4.5.2.e regarding 18-month surveillances for emergency core cooling subsystems. The proposed change would add components to the surveillances by including the piping outside containment which is in contact with sump water during loss-of-coolant accident conditions.

(iv) Items 159 and 160 deal with Specification 4.8.1.3 regarding surveillance requirements for the cathodic protection system. The proposed changes would modify the surveillance intervals from 92 to 61 days and from 18 to 12 months.

(v) Item 207 deals with Specification 6.8.1.g regarding limitations on making modifications to the core protection calculator software. The proposed change would include additional limitations on making modifications to the software. Item 80 is an associated administrative change to the bases section for the core protection calculator to make it consistent with the proposed change to Specification 6.8.1.g.

(c) The remaining items of proposed changes are either editorial, provide clarification, remove redundancy or correct errors. Examples of these types are as follows:

(i) Item 203 deals with Specification 6.5.3.5.1 regarding the audit of the Pre-planned Alternate Sampling Program (PASP) and its implementing procedures. The proposed change would delete this redundant Specification since the controls for PASP and its procedures are addressed in Specification 6.8.1. Item 211 deals with Specification 6.16 regarding (1) NRC approval of PASP by Region V prior to implementation and (2) reporting changes to PASP in the Semiannual Radioactive Effluent Release Report. The proposed change would delete this Specification since (1) NRC approval for PASP was granted on January 14, 1986 and (2) Specification 6.9.1.8, which defines the content for the Semiannual Radioactive Effluent Release Report, doesn't require that changes to PASP be reported therein.

(ii) Item 92 deals with the footnotes for Specifications 3.4.1.4.1 regarding operability of the shutdown cooling loops. The proposed editorial change would move one footnote ahead of the other two since it appears first in the text.

(iii) Item 115 deals with Specification 3.4.10 regarding reactor coolant system vent paths. The proposed change would clarify the locations for verifying the operability of the reactor coolant system vent paths by specifically stating that the locations are the reactor vessel head and the pressurizer steam space.

(iv) Item 155 deals with a misspelled word on page 3/4 7-22. The proposed change would correctly spell the word "susceptible."

3.0 EVALUATION

The staff has reviewed the above changes. As a result of that review, the staff has made the following determinations.

Essentially all of the proposed changes have no safety significance and are administrative in nature in that they are either editorial, provide clarification, remove redundancy or correct errors. The remaining few are changes which respond to changes in regulation or regulatory guidance, or are changes which are more restrictive than the current Technical Specifications. All of the areas of change were previously reviewed and accepted by the staff in developing the Technical Specifications for Palo Verde, Units 2 and 3, and are included in the Specifications issued for Palo Verde, Units 2 and 3. All these changes will make those portions of the Palo Verde, Unit 1 Specifications consistent with the Specifications for Units 2 and 3.

On the basis of the above evaluation, the staff finds the proposed changes to the Palo Verde, Unit 1 Specifications to be acceptable.

4.0 CONTACT WITH STATE OFFICIAL

The Arizona Radiation Regulatory Agency has been advised of the proposed determination of no significant hazards consideration with regard to these changes. No comments were received.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves administrative changes. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public. We, therefore, conclude that the proposed changes are acceptable.

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Dated: March 2, 1988