TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION	
HV-9339	Shutdown cooling flow from reactor coolant loop 2	Permanently Bypassed
HV-9340	SI tank T008 to reactor coolant loop 1A	Permanently Bypassed
HV-9370	SI tank T010 to reactor coolant loop 2B	Permanently Bypassed
HV-9347	SI pump minimum recirculation	Permanently Bypassed
HV-9322	LPSI to reactor coolant loop 1A	Permanently Bypassed
HV-9331	LPSI to reactor coolant loop 2B	Permanently Bypassed
-9348	SI pump minimum recirculation	Permanently Bypassed
HV-9323	HPSI to reactor coolant loop 1A	Permanently Bypassed
HV-9332	HPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9217	RCP bleed off to volume control tank	Permanently Bypassed
HV-9326	HPSI to reactor coolant loop 1B	Permanently Bypassed
HV-9329	HPSI to reactor coolant loop 2A	Permanently Bypassed
HV-7258	Waste gas header containment isolation	Permanently Bypassed
HV-0508	Reactor coolant hot leg sample containment isolation	Permanently Bypassed
HV-0517	Reactor coolant hot leg sample containment isolation	Permanently Bypassed
HV-9368	Shutdown HX to containment spray	Permanently Bypassed
HV-0510	Pressurizer vapor sample containment isolation	Permanently Bypassed
HV-0512	Pressurizer surge line liquid sample containment isolation	Permanently Bypassed
HV-9950	Containment purge outlet	Permanently Bypassed
HV-9917	Hydrogen purge exhaust inlet	Permanently Bypassed
-9946	Hydrogen purge supply discharge	Permanently Bypassed

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TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	
HV-9302	Containment emergency sump outlet	Permanently Bypassed
HV-9304	Containment emergency sump outlet	Permanently Bypassed
HV-6211	CCW to containment	Permanently Bypassed
HV-6368	CCW to emergency cooling unit	Permanently Bypassed
HV-6369	CCW from emergency cooling unit	Permanently Bypassed
HV-6216	CCW from containment	Permanently Bypassed
HV-6372	CCW to emergency cooling unit	Permanently Bypassed
HV-6373	CCW from emergency cooling unit	Permanently Bypassed
HV-9900	Containment normal cooling supply isolation	Permanently Bypassed
HV-9971	Containment normal cooling return isolation	Permanently Bypassed
LV-0227	Boric Acid makeup control	Permanently Bypassed
HV-4713	Aux. F.W. to steam generator control	Permanently Bypassed
HV-9334	SI tank drain to refueling water tank	Permanently Bypassed
HV-9350	SI tank T007 to reactor coolant loop 1B	Permanently Bypassed
HV-9360	SI tank T009 to reactor coolant loop 2A	Permanently Bypassed
HV-9325	LPSI to reactor coolant loop 1B	Permanently Bypassed
HV-9328	LPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9201	Aux. spray to pressurize	Permanently Bypassed
HV-9327	HPSI to reactor coolant loop 1B	Permanently Bypassed
HV-9330	HPSI to reactor coolant loop 2A	Permanently Bypassed
HV-6223	CCW Non-Crit Containment inlet isolation	Permanently Bypassed
HV-9324	HPSI to reactor coolant loop 1A	Permanently Bypassed
HV-9333	HPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9337	Shutdown coolant flow from reactor coolant loop 2	Permanently Bypassed

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	
HV-0516	Reactor coolant drain tank sample containment isolation	Permanently Bypassed
HV-7512	Containment isolation reactor coolant drain to R.W. system	Permanently Bypassed
HV-9367	Shutdown HX to containment spray header	Permanently Bypassed
HV-0514	Quench tank vapor sample containment isolation	Permanently Bypassed
HV-5803	Containment sump to R.W. sump	Permanently Bypassed
HV-9949	Containment purge inlet	Permanently Bypassed
HV-9303	Containment emergency sump outlet	Permanently Bypassed
HV-9305	Containment emergency sump outlet	Permanently Bypassed
HV-6366	CCW to emergency cooling unit	Permanently Bypassed
HV-6367	CCW from emergency cooling unit	Permanently Bypassed
HV-6236	CCW Non-crit. containment outlet isolation valve	Permanently Bypassed
HV-6370	CCW to emergency cooling unit	Permanently Bypassed
HV-6371	CCW from emergency cooling unit	Permanently Bypassed
HV-8150	Shutdown cooling HX outlet	Permanently Bypassed
HV-8151	Shutdown cooling HX outlet	Permanently Bypassed
HV-9306	SI pump mini-flow	Permanently Bypassed
HV-9307	SI pump mini-flow	Permanently Bypassed
HV-9247	Boric acid pumps to charging pump suction	Permanently Bypassed
HV-9379	Shutdown cooling flow to LPSI	Permanently Bypassed
HV-9353	Shutdown cooling warm up valve	Permanently Bypassed
HV-9420	HPSI to reactor coolant loop 2	Permanently Bypassed
HV-6497	Saltwater from CCW HX	Permanently Bypassed
HV-9300	Refueling water tank east outlet	Permanently Bypassed

TABLE 3.8-2 (Continued)

VALVE NUMBER	FUNCTION	
HV-5686	Firewater to containment isolation	Permanently Bypassed
HV-0227B	Volume control tank drain return	Permanently Bypassed
HV-9240	Boric acid makeup tank T072 to charging pump suction	Permanently Bypassed
HV-9336	Shutdown cooling flow to LPSI pump suction	Permanently Bypassed
HV-9359	Shutdown cooling warm up valve	Permanently Bypassed
HV-9301	Refueling water tank west outlet	Permanently Bypassed
HV-6495	Saltwater from CCW HX	Permanently Bypassed
TV-9267	Reactor coolant regenerative HX isolation	Permanently Bypassed
HV-9434	HPSI to reactor coolant loop 1 hot leg	Permanently Bypassed
HV-8151	Reactor aux. shutdown cooling HX outlet	Permanently Bypassed
HV-8153	Reactor aux. shutdown cooling HX outlet	Permanently Bypassed
HV-4712	Aux F.W. Steam gen. control	Permanently Bypassed

ATTACHMENT "B"

TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD

PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION	
HV-9339	Shutdown cooling flow from reactor coolant loop 2	Permanently Bypassed
HV-9340	SI tank T008 to reactor coolant loop 1A	Permanently Bypassed
HV-9370	SI tank T010 to reactor coolant loop 2B	Permanently Bypassed
HV-9347	SI pump minimum recirculation	Permanently Bypassed
HV-9322	LPSI header to reactor coolant loop 1A	Permanently Bypassed
HV-9331	LPSI header to recator coolant loop 2B	Permanently Bypassed
HV-9348	SI pump minimum recirculation	Permanently Bypassed
HV-9323	HPSI header #2 to reactor coolant loop 1A	Permanently Bypassed
HV-9332	HPSI header #2 to reactor coolant loop 2B	Permanently Bypassed
HV-9217	RCP bleed off to volume control tank-cont. isol.	Permanently Bypassed
HV-9326	HPSI header #2 to reactor coolant loop 1B	Permanently Bypassed
HV-9329	HPSI header #2 to reactor coolant loop 2A	Permanently Bypassed
HV-7258	Waste gas surge tank header containment isolation	Permanently Bypassed
HV-0508	Reactor coolant hot leg #1 sample containment isolation	Permanently Bypassed
HV-0517	Reactor coolant hot leg #2 sample containment isolation	Permanently Bypassed
HV-9358	Shutdown HX E003 to containment spray header #2	Permanently Bypassed
HV-0510	Pressurizer vapor sample containment isolation	Permanently Bypassed
HV-0512	Pressurizer surge line liquid sample containment isolation	Permanently Bypassed

TABLE 3.8-2 (continued)

HV-9950	Containment purge outlet to exhaust Unit A060 containment isolation	Permanently Bypassed
HV-9917	Hydrogen purge exhaust Unit A082 inlet containment isolation	Permanently Bypassed
HV-9946	Hydrogen purge supply Unit A080 discharge containment isolation	Permanently Bypassed
HV-9302	Containment emergency sump outlet	Permanently Bypassed
HV-9304	Containment emergency sump outlet	Permanently Bypassed
HV-6211	CCW non-critical loop to containment isolation valve	Permanently Bypassed
HV-6368	CCW loop B to emergency cooling unit E400	Permanently Bypassed
HV-6369	CCW from emergency cooling unit E400 to loop B	Permanently Bypassed
HV-6216	CCW non-critical loop from containment isolation valve	Permanently Bypassed
HV-6372	CCW loop B emergency cooling unit E402	Permanently Bypassed
HV-6373	CCW loop B from emergency cooling unit E402	Permanently Bypassed
HV-9900	Containment normal cooling suppl1y isolation	Permanently Bypassed
HV-9971	Containment normal cooling return isolation	Permanently Bypassed
LV-0227C	Boric Acid makeup control	Permanently Bypassed
HV-4713	Aux. F.W. P141 discharge to steam generator E089 control valve	Permanently Bypassed
HV-9334	SI tank drain header to refueling water tank containment isolation	Permanently Bypassed
HV-9350	SI tank T007 to reactor coolant loop 1B	Permanently Bypassed
HV-9360	SI tank T009 to reactor coolant loop 2A	Permanently Bypassed
HV-9325	LPSI header to reactor coolant loop 1B	Permanently Bypassed
HV-9328	LPSI header to rector coolant loop 2A	Permanently Bypassed
HV-9201	Aux. spray to pressurizer	Permanently Bypassed

TABLE 3.8-2 (continued)

HV-9327	HPSI header #1 to reactor coolant loop 1B	Permanently Bypassed
HV-9330	HPSI header #1 to reactor coolant loop 2A	Permanently Bypassed
HV-6223	CCW non-critical loop containment inlet isolation	Permanently Bypassed
HV-9324	HPSI header #1 reactor coolant loop 1A	Permanently Bypassed
HV-9333	HPSI header #1 to reactor coolant loop 2B	Permanently Bypassed
HV-9337	Shutdown coolant flow from reactor coolant loop 2	Permanently Bypassed
HV-9377	Shutdown coolant flow from reactor coolant loop 2	Permanently Bypassed
HV-9378	Shutdown coolant flow from reactor coolant loop 2	Permanently Bypassed
HV-0516	Reactor coolant drain tank sample containment isolation	Permanently Bypassed
HV-7512	Containment isolation reactor coolant drain tank to R.W. system	Permanently Bypassed
HV-9367	Shutdown HX E004 to containment spray header #1	Permanently Bypassed
HV-0514	Quench tank vapor sample containment isolation	Permanently Bypassed
HV-5803	Containment sump to R.W. sump	Permanently Bypassed
HV-9949	Containment purge inlet from supply unit A374 isolation	Permanently Bypassed
HV-9303	Containment emergency sump outlet	Permanently Bypassed
HV-9305	Containment emergency sump outlet	Permanently Bypassed
HV-6366	CCW loop A to emergency cooling unit E401	Permanently Bypassed
HV-6367	CCW loop A from emergency cooling unit E401	Permanently Bypassed
HV-6236	CCW non-critical containment outlet isolation valve	Permanently Bypassed
HV-6370	CCW loop A to emergency cooling unit E399	Permanently Bypassed
HV-6371	CCW loop A from emergency cooling unit E399	Permanently Bypassed
HV-8150	Shutdown cooling HX E004 outlet isolation valve	Permanently Bypassed

TABLE 3.8-2 (continued)

HV-8151	Shutdown cooling HX E003 outlet isolation valve	Permanently Bypassed
HV-9306	SI pump minimum recirculation	Permanently Bypassed
HV-9307	SI pump minimum recirculation	Permanenlty Bypassed
HV-9247	Boric acid pumps to CVC charging pump suction	Permanently Bypassed
HV-9379	Shutdown cooling flow to LPSI	Permanently Bypassed
HV-9353	Shutdown cooling warm-up valve	Permanently Bypassed
HV-9420	HPSI header #1 to reactor coolant loop 2 hot leg	Permanently Bypassed
HV-6497	Saltwater from CCW HX E001	Permanently Bypassed
HV-9300	Refueling water tank east T005 outlet	Permanently Bypassed
HV-5686	Firewater to containment isolation	Permanently Bypassed
HV-0227B	Volume control tank T077 drain return	Permanently Bypassed
HV-9240	Boric acid makeup tank T071 to charging pump	Permanently Bypassed
HV-9235	Boric acid makeup tank T072 to charging pump suction	Permanently Bypassed
HV-9336	Shutdown cooling flow to LPSI pump suction	Permanently Bypassed
HV-9359	Shutdown cooling warm up valve	Permanently Bypassed
HV-9301	Refueling water tank west T-006 outlet	Permanently Bypassed
HV-6495	Saltwater from CCW HX E002	Permanently Bypassed
HV-9267	Letdown line containment isolation valve	Permanently Bypassed
HV-9434	HPSI header #2 to reactor coolant loop 1 hot leg	Permanently Bypassed
HV-8152	Shutdown cooling HX inlet isolation valve	Permanently Bypassed
HV-8153	Shutdown cooling HX inlet isolation valve	Permanently Bypassed
HV-4712	Aux. F.W. pump P504 discharge to steam generator control	Permanently Bypassed

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DESCRIPTION OF PROPOSED CHANGE NPF-10-7 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 3 OPERATING LICENSE NPF-10

This is a request to add a new section, Section 7.0, to Appendix "A" Technical Specification.

7.0 SPECIAL TEST PROGRAM

Existing Specification

None

Proposed Specifications

See Attachment "A"

Section 7.0 is to be added to the Technical Specifications for the duration of the special low power test program as described in Section I.G.1 of the Safety Evaluation Report (SER), NUREG-0712 Supplement No. 1.

Reason For Proposed Change

By letter dated February 2, 1982, SCE committed to a natural circulation test program which conforms to the current NRC staff position. Certain of these tests require the plant to be in configuration outside of the normal envelope defined by the Technical Specifications. This new Section 7.0 consists of a table of technical specifications which will be considered not applicable for the duration of the specified test. As a compensatory measure, the tests will be terminated if any of several parameters clearly identified in the procedures is exceeded.

Safety Analysis

The safety analysis for this Proposed Change is contained in CEN-201(S) Natural Circulation Test Program, San Onofre Nuclear Generating Station, Unit 2 Safety Evaluation, April 1982. This Safety Evaluation was transmitted with the applicable procedures on April 15, 1982.

Based on the results of this evaluation, it is concluded that: (1) Proposed Change NPF-10-7 does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

7.0 SPECIAL TEST PROGRAM

7.1 For conducting the special low power test program as described in Section I.G.1 of the Safety Evaluation Report (SER) the Technical Specifications may be exempt as follows:

Technical Specifications Section	Description	SER A1	SER A2	SER A3	SER B1&B2	SER B3
2.1.1	Safety Limits - Reactor Core	X	X	X		
2.2.1	Reactor Trip Setpoints					
3.	Logarithimic Power Level-High	X	X	X		
5.	Pressurizer Pressure-Low		X			
7.	Steam Gen. Pressure-Low	X	X	X		
9.	Local Power Density-High	X	X	X		
10.	DNBR-Low	X	X	X		
11.	Reactor Coolant Flow-Low	X	X	X		
3.1.1.4	Minimum Temperature for Criticality	X	X	X		
3.3.1	Reactor Protective Instrumentation					
3.	Logarithmic Power Level-High	X	X	X		
9.	Local Power Density-High	X	X	X		
10.	DNBR-Low	X	X	X		
14.	Core Protection Calculators	X	X	X		
16.	Reactor Coolant Flow-Low	X	X	X		
3.3.2	Engineered Safety Feature Actuation System Instrumentation					
1.	Safety Injection (SIAS)		X			
4.	Main Steam Line Isolation	X	X	X		
6.	Containment Cooling (CCAS)		X			
8.	Emergency Feedwater (EFAS)	X	X	X		
3.4.1	Reactor Coolant Loops and Coolant Circulation	X	X	X		
3.4.1.2	Hot Standby	X	X	X	X	X
3.4.3	Pressurizer		X		X	
3.4.4	Steam Generators			X		
3.7.1.2	Auxiliary Feedwater System		X	X	X	

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DESCRIPTION OF PROPOSED CHANGE NPF-10-10 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 3 OPERATING LICENSE NPF-10

This is a request to make various editorial and typographical changes and resolve several inconsistencies in Appendix "A" Technical Specifications.

Existing Specifications

See Attachment "A"

Proposed Specifications

The proposed specifications are as follows and contained in Attachment "B" if noted:

Page	
3/4 1-12 and 3/4 1-14	Technical Specifications 3.1.2.7.b.3 and 3.1.2.8.b.3 should be in conformance with Tech. Spec. 3.5.4.c and therefore the upper temperature limit should be changed from 120°F to 100°F. Consistent with this reasoning, Surveillance Requirement 4.1.2.7.b and 4.1.2.8.b should read "outside air temperature is less than 40°F or greater than 100°F." to be in conformance with Surveillance Requirement 4.5.4.b. The more restrictive value of 100°F should be reflected in the Technical Specifications.
3/4 1-25	Technical Specification 3.1.3.7, contained in Attachment "B", is rewritten to allow the part length CEA group to be withdrawn to $\geq$ 145" and left full out without the necessity of withdrawing the part length CEA group to the upper electrical limit. This change will permit Specification 3.1.3.7 to be in conformance with Specifications 3.1.3.4 and 3.1.3.5 which defines fully withdrawn as $\geq$ 145.
3/4 3-4	Technical Specfication 3.3.1, Table 3.3-1, notation (c) should read "bypass shall be automatically removed when THERMAL POWER is greater than or equal to 5% of RATED THERMAL POWER" not 1%, to be consistent with the actual operation of the bistable. Table 3.3-1 ACTION 2 Line 7 reads, "Specification 6.5.1.6k" and should read "6.5.1.6e" because parts of Section 6.5.1.6 were deleted and item "k" was relettered as item "e".
3/4 3-14	Technical Specification 3.3.2, Table 3.3-3 item 2, Containment Spray (CSAS) is required to be operable in Modes 1, 2, and 3 but not Mode 4 as defined in Technical Specification 3.6.2.1. The requirement of Mode 4 should therefore be deleted.

Page

3/4 3-15 Technical Specification 3.3.2, Table 3.3-3 Item 5 is inconsistent with the implied operability requirement Technical Specification 3.5.3 which recognizes the additional applicable Mode 4. Therefore, Mode 4 should be added to Item 5 Table 3.3-3 under Applicable Modes.

3/4 3-19 Technical Specification 3.3.2, Table 3.3-3 notation "a" should state "bypass shall be automatically removed when pressurizer pressure is greater than or equal to 400 psia", not 500 psia. This change will be consistent with FSAR Section 7.2.1.1.1.6 and Technical Specification 3.3.1, Table 3.3-1 notation "b".

3/4 3-22 Technical Specification 3.3.2, Table 3.3-4 Item 5a Manual (RAS) should be deleted. There are no manual RAS (Trip Buttons) in the plant and therefore should be deleted. Reletter Item 5 as applicable.

3/4 3-29 and 3/4 3-30 Technical Specification 3.3.2, Table 3.3-5 Items 8 and 9 Auxiliary Feedwater (AC trains), change the response time on both items from "40.9*" to "50.9*/40.9**". The 40.9 second requirement pertains to non-LOCA events which include EDG starting in SIAS and pump load sequence delay. Such events are bounded for AFW delivery time by the loss of normal feedwater event and require AFW delivery in 42.7 seconds (40.9 is conservative). Events which require AFW when SIAS is present (e.g., small break LOCA) are bounded for AFW delivery time by the (coincident) loss of normal A/C event and require AFW delivery in 53 seconds (50.9 is conservative). This change therefore makes the applicability of AFW pump load sequence delay to EFAS response time consistent with design basis requirements. Due to this change the following will need to be added to the bottom of page 3/4 3-30 "** Emergency diesel generator starting delay (10 seconds) is included".

3/4 3-53 Technical Specification 3.3.3.6, Table 3.3-10, change item 23 Cold Leg HPSI Flow from 2/cold leg to 1/cold leg for the required number of channels and 1/cold leg to N.A. under minimum channels operable. The plant has 4 cold legs and one HPSI flow channel per cold leg. The proposed change is contained in Attachment "B".

3/4 4-3 Technical Specification 3.4.1.3 specifies:

#With the Reactor Coolant System cold leg temperature less than or equal to 235°F, the SDCS isolation valves HV-9337, HV-9339, HV-9377, and HV-9378 shall be open with the SDCS relief valve PSV-9349 OPERABLE.

Surveillance Requirement 4.4.8.3.1.1.a (Page 3/4 4-32) specifies:

a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that at least one pair of SDCS Relief Valve isolation valves (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) is open.

The statement in Spec. 3.4.1.3 is a SURVEILLANCE REQUIREMENT contained in an APPLICABILITY statement. This is inconsistent and redundant to Surveillance Requirement of 4.4.8.3.1.1.a and should be deleted.

3/4 4-5 Technical Specification 3.4.1.4.1, should be rewritten as "At least one shutdown cooling train shall be OPERABLE and in operation* either:" The mention of suction line valves to be opened should be deleted from the first two lines. These valves are controlled by the low temperature overpressure protection Technical Specification 3.4.8.3.1, ACTION item "b" which allows some of the SDCS valves to be closed at times.

3/4 4-8 Technical Specifications 4.4.3.2 should have the words "from the 1E busses." added to the end of the sentence. This added information is for clarification as required by SER II.E.3.1.

3/4 4-32 Technical Specification 3.4.8.3.1 APPLICABILITY needs changed from "one any" to "any one".

3/4 7-9 Technical Specification 3.7.1.5, Modes 2 and 3, The word "in" was inadvertently omitted. It should read "Otherwise, be in at least HOT STANDBY within the next 6 hours...."

3/4 10-6 Technical Specification 3.10.5, Table 3.10-1, item 2.a Main Steam Line Area Monitor "2RT-7847B1" should be "2RT-7874B1".

3/4 10-8 Technical Specification 3.10.6, line 1, the word "of" is misspelled as "fo" and requires correction.

B 3/4 7-7 Fire Suppression Systems, insert new paragraph.

"The San Onofre Unit 2&3 fire pumps and water supplies, supply water to the San Onofre Unit 1 fire system. Satisfactory completion of the Unit 2&3 fire pump and water supply surveillance requirements, automatically satisfies the Unit 1 fire water supply requirements." This will clarify the relationship between the two fire systems.

B 3/4 8-2 Surveillance Requirement 4.8.1.1.2.c requires obtaining fuel oil samples in accordance with ASTM-D270-1975. As indicated in Part 23 of the ASTM manual, the appropriate standard for obtaining these samples is ASTM-D270-1965 (Reverified 1975). However Reg. Guide 1.137 paragraph C.2.c specifically calls for ASTM-D270-1975. Therefore, for clarity the following should be added to the end of the second paragraph of the BASES, page B 3/4 8-2: "Reg. Guide 1.137 recommends testing of fuel oil samples in accordance with ASTM-D270-1975 however ASTM-D270-1965 was reverified in 1975 and is therefore the appropriate standard to be used".

Reason for Proposed Change

The various corrections contained in this proposed change are for clarification only.

Safety Analysis

Corrections contained in this Proposed Change NPF-10-10 are editorial or typographical and do not change the intent of the Technical Specifications.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-10 does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

HP:4082

ATTACHMENT A

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.7 As a minimum, one of the following borated water sources shall be OPERABLE:

a. One boric acid makeup tank and at least one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water storage tanks with:

1. A minimum borated water volume of 5465 gallons above the ECCS suction connection,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 120°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

2. Verifying the contained borated water volume of the tank, and

3. Verifying the boric acid makeup tank solution temperature when it is the source of borated water.

b. At least once per 24 hours by verifying the RWST temperature when it is the source of borated water when the outside air temperature is less than 40°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. At least one boric acid makeup tank and at least one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water storage tank with:

   1. A minimum contained borated water volume of 362,800 gallons above the ECCS suction connection,
   2. Between 1720 and 2300 ppm of boron, and
   3. A solution temperature between 40°F and 120°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2% delta k/k at 200°F; restore the above required boric acid makeup tank 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 one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration in the water,
   2. Verifying the contained borated water volume of the water source, and
   3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is less than 40°F.

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

TABLE 3.3-1 (Continued)

TABLE NOTATION

@To be OPERABLE prior to first exceeding 5% RATED THERMAL POWER.

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

#The provisions of Specification 3.0.4 are not applicable.

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

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

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

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

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

(g) Trip may be bypassed below 55% RATED THERMAL POWER.

ACTION STATEMENTS

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

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sest of 2	1, 2, 3, 4	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9*, 10*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	9*, 10*
d. Automatic Actuation - Logic	4	2	3	1, 2, 3, 4	9*, 10*
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	9*, 10*
c. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	9*, 10*
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
b. Manual SIAS (Trip Buttons) (c)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
c. Containment Pressure - High	4	2	3	1, 2, 3	9*, 10*
d. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	9*, 10*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3	11
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	9*, 10*
c. Automatic Actuation Logic	4/steam generator	2/steam generator	3/steam generator	1, 2, 3	9*, 10*
5. RECIRCULATION (RAS)					
a. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	9*, 10*
b. Automatic Actuation Logic	4	2	3	1, 2, 3	9*, 10*
6. CONTAINMENT COOLING (CCAS)					
a. Manual CCAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	8
b. Manual SIAS (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3	8
c. Automatic Actuation Logic	4	2	3	1, 2, 3, 4	9*, 10*

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 400 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 500 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Actuated equipment only; does not result in CIAS.

The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

** With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 8 - 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 9 - 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 6.5.1.6k. 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	Functional Unit Bypassed
1.	Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)
3.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
c. Pressurizer Pressure - Low	$\geq$ 1806 psia (1)	$\geq$ 1763 psia (1)
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	$\leq$ 16.14 psig	$\leq$ 16.83 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)(5)	Not Applicable	Not Applicable
c. Containment Pressure - High	$\leq$ 2.95 psig	$\leq$ 3.14 psig
d. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 729 psia (2)	$\geq$ 711 psia (2)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
5. RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank	18.5% of tap span	19.27% $\geq$ tap span $\geq$ 17.73%
c. Automatic Actuation Logic	Not Applicable	Not Applicable

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. Steam Generator Pressure - Low	
a. MSIS	
(1) Main Steam Isolation (MSIV)	20.9
(2) Main Feedwater Isolation	10.9
6. Refueling Water Storage Tank - Low	
a. RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Valves Shut	40.7*
7. 4.16 kv Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low (and No Pressure-Low Trip)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	40.9*
(2) Auxiliary Feedwater (steam/DC train)	30.9
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	40.9*
(2) Auxiliary Feedwater (Steam/DC train)	30.9
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

Table 3.3-5 (Continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
13. Control Room Toxic Gas (Butane/Propane)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)
14. Control Room Toxic Gas (Carbon Dioxide)	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)
15. Fuel Handling Building Airborne Radiation	
a. FHIS	
(1) Fuel Handling Building Post-Accident Cleanup Filter System	Not Applicable
16. Containment Airborne Radiation	
a. CPIS	
(1) Containment Purge Isolation	2 (NOTE 2)
17. Containment Area Radiation	
a. CPIS	
(1) Containment Purge Isolation	2 (NOTE 2)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.

* Emergency diesel generator starting delay (10 sec.) and sequence loading delays for SIAS are included.

2. Response time includes emergency diesel generator starting delay (applicable to AC motor operated valves other than containment purge valves), instrumentation and logic response only. Refer to table 3.6-1 for containment isolation valve closure times.

3. All CIAS-Actuated valves except MSIVs and MFIVs.

4. CCW non-critical loop isolation valves 2HV-6212, 2HV-6213, 2HV-6218 and 2HV-6219.

5. Response time includes instrumentation, logic, and isolation damper closure times only.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT		REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
19.	Containment Area Radiation - High Range	2	1
20.	Main Steam Line Area Radiation	1/steam line	N.A.
21.	Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.
22.	Purge/Vent Stack Radiation Monitor - Wide Range*	2	1
23.	Cold Leg HPSI Flow	2/cold leg	1/cold leg
24.	Hot Leg HPSI FLow	1/hot leg	N.A.

NOTES:

*The two required channels are the Unit 2 monitor and the Unit 3 monitor.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the loop(s)/train(s) listed below shall be OPERABLE:

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

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

3. Shutdown Cooling Train A,

4. Shutdown Cooling Train B.

b. At least one of the above Reactor Coolant loops and/or shutdown cooling trains shall be in operation.*

APPLICABILITY: MODE 4#

ACTION:

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

b. With no Reactor Coolant loop or shutdown cooling train 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/ train to operation.

#With the Reactor Coolant System cold leg temperature less than or equal to 235°F, the SDCS isolation valves HV-9337, HV-9339, HV-9377, and HV-9378 shall be open with the SDCS relief valve PSV-9349 OPERABLE.

*All Reactor Coolant pumps and shutdown cooling pumps 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.

**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 235°F unless 1) the pressurizer water volume is less than 900 cubic feet or 2) the secondary water temperature 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 FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling train shall be OPERABLE with all suction line valves open and in operation,* and either:

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

b. The secondary side water level of each steam generator shall be greater than 10% (wide range).

APPLICABILITY: MODE 5#, with Reactor Coolant loops filled.

ACTION:

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

b. With no shutdown cooling train 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 train to operation.

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.1.2 The shutdown cooling train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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

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

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 900 cubic feet and at least two groups of pressurizer heaters each having a capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, 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, 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 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually energizing the heaters.

4.4.3.3 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

RCS TEMPERATURE $\leq$ 235°F

LIMITING CONDITION FOR OPERATION

3.4.8.3.1 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System (SDCS) Relief Valve (PSV9349) with a lift setting of less than or equal to 402 psig, or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 5.6 square inches.

APPLICABILITY: MODE 4 when the temperature of one any RCS cold leg is less than or equal to 235°F; Mode 5; Mode 6 with the reactor vessel head on.

ACTION:

a. With the SDCS Relief Valve inoperable, reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 square inch vent within the next 8 hours.

b. With one or both SDCS Relief Valve isolation valves in a single SDCS Relief Valve isolation valve pair (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) closed, open the closed valve(s) within 7 days or reduce T_{avg} to less than 200°F, depressurize and vent the RCS through a greater than or equal to 5.6 inch vent within the next 8 hours.

c. In the event either the SDCS Relief Valve or an RCS vent is used to mitigate a 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 SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

a. Verifying at least once per 72 hours when the SDCS Relief Valve is being used for overpressure protection that at least one pair of SDCS Relief Valve isolation valves (valve pair 2HV9337 and 2HV9339 or valve pair 2HV9377 and 2HV9378) is open.

PLANT SYSTEMS

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 and 3.

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, reduce power to less than or equal to 5 percent RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2 or 3 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 with the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5.0 seconds when tested pursuant to Specification 4.0.5.

TABLE 3.10-1

RADIATION MONITORING/SAMPLING EXCEPTIONS

1. Testing performed pursuant to FSAR Section 11.5.2.1.5.2 in startup program shall satisfy the initial CHANNEL CALIBRATION for the following monitors prior to first exceeding 5% RATED THERMAL POWER:

a.	Control Room Airborne Monitors	2RT-7824 2RT-7825
b.	Containment Airborne Monitors	2RT-7804-1 2RT-7807-2
c.	Containment Purge Area Monitors	2RT-7856-1 2RT-7857-2
d.	Containment Area Radiation - High Range Monitors	2RT-7820-1 2RT-7820-2
e.	Plant Vent Stack Airborne Monitor	2/3RT-7808
f.	Radwaste Discharge Line Monitor	2/3RT-7813
g.	Blowdown Neutralization Sump Monitor	2RT-7817
h.	Turbine Building Sump Monitor	2RT-7821

2. The following monitors and samplers shall be OPERABLE prior to first exceeding 5% RATED THERMAL POWER:

a.	Main Steam Line Area Monitors	2RT-7874A1 2RT-7847B1 2RT-7875A1 2RT-7875B1
b.	Condenser Evacuation System - Wide Range Monitor	2RT-7870-1
c.	Purge/Vent Stack Monitors - Wide Range	2RT-7865-1 3RT-7865-1
d.	Plant Vent Stack	Flow Rate Monitor
e.	Containment Purge	Flow Rate Monitor
f.	Condenser Evacuation System	Iodine Sampler Particulate Sampler Flow Rate Monitor

3. The Steam Jet Air Ejector Monitor (2RT-7818) shall be OPERABLE prior to initial criticality.

SPECIAL TEST EXCEPTIONS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.6 The minimum temperature for criticality limits fo Specification 3.1.1.4 and the MODE 2 definition of Table 1.1 may be suspended during low temperature PHYSICS TESTS to a minimum temperature of 320°F provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER.

b. The reactor trip setpoints on the OPERABLE Linear Power Level - High neutron flux monitoring channels are set at ≤ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship and the minimum temperaure for criticaliaty is maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: MODE 2.*

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately trip the reactor.

b. With the Reactor Coolant System temperature and pressure relationship and/or the minimum temperature for criticality within the region of unacceptable operation on Figure 3.4-2, immediately trip the reactor and, if necessary, restore the temperature-pressure relationship to within its limit within 30 minutes; perform the engineering evaluation required by Specification 3.4.8.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.6.1 The Reactor Coolant System temperature and pressure relationship and the minimum temperature for criticality shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.6.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.

4.10.6.3 The Reactor Coolant System temperature shall be verified to be greater than or equal to 320°F at least once per hour.

4.10.6.4 Each Logarithmic Power Level and Linear Power Level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

*First core only, prior to first exceeding 5% RATED THERMAL POWER.

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.9 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensure that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire windows, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

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

as well as operation of loss of voltage logic, is the same as for the primary connection using the reserve auxiliary transformer, with the exception of no transfer to the companion unit.

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

Additionally, Regulatory Guide 1.9 allows loading of the diesel generator to its 2000 hour rating in an accident situation. The full load, continuous operation rating for each diesel generator is 4700 kw, while the calculated accident loading is 4000 kw. No 2000 hour loading has been specified by the diesel generator manufacturer and, as a result the full loading rating of 4700 kw is conservatively established as the 2000 hour rating. Diesel frequency droop restrictions are established due to HPSI flow rate considerations.

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 onfloat 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 electrolye level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 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 .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

ATTACHMENT B

REACTIVITY CONTROL SYSTEMS

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be:

a. withdrawn to $\geq$ 145" or;

b. restricted to prevent the neutron absorber section of the part length CEA group from covering the same axial segment ($\leq$ 145") of the fuel assemblies for a period in excess of 7 EFPD out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2

ACTION:

With the neutron absorber section of the part length CEA group covering any same axial segment of the fuel assemblies as specified in 3.1.3.7.b above, for a period exceeding 7 EFPD out of any 30 EFPD period, either:

a. Reposition the part length CEA group to ensure no neutron absorber section of the part length CEA group is covering the same axial segment of the fuel assemblies within 2 hours, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
19. Containment Area Radiation - High Range	2	1
20. Main Steam Line Area Radiation	1/steam line	N.A.
21. Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.
22. Purge/Vent Stack Radiation Monitor - Wide Range*	2	1
23. Cold Leg HPSI Flow	1	N.A.
24. Hot Leg HPSI Flow	1/hot leg	N.A.

NOTES:

*The two required channels are the Unit 2 monitor and the Unit 3 monitor.

DESCRIPTION OF PROPOSED CHANGE NPF-10-11
AND SAFETY ANALYSIS AMENDMENT APPLICATION NO. 3
OPERATING LICENSE NPF-10

This is a request to revise Appendix A Technical Specification 6.1.2, 6.2.2, 6.5.2, and 6.8.2.

Technical Review and Control

Reason for Proposed Change

As written, the Technical Specifications 6.5.2 and 6.8.2 require "Station Supervisory Personnel" to review all procedures which affect "Nuclear Safety". This is an excessive burden on station supervisory personnel when 10 CFR 50, Appendix B allows; "The Applicant may delegate to others such as contractors, agents or consultants, the work of establishing and executing the quality assurance program or any part thereof, but shall retain responsibility therefore." We, therefore, wish to modify the Technical Specifications to allow certain specified review and audit functions to be performed by organizations other than station supervisory personnel. In addition, Technical Specifications 6.1.2, 6.2.2, Table 6.2-1, Figure 6.2-2 and Figure 6.2-3, are updated to reflect current personnel titles.

Existing Specifications

See Attachment "A".

Proposed Specification

Proposed change to Technical Specification 6.2.2, Figure 6.2-2 is contained in Attachment "B". Technical Specifications 6.1.2, 6.2.2.e, Table 6.2-1 and Figure 6.2-3 are changed to "Shift Supervisor" from "Watch Engineer." The proposed change to the applicable paragraphs of Technical Specifications 6.5.2, 6.5.3, and 6.8.2 are as follows:

6.5.2.1 Each procedure and program required by Specification 6.8 and changes thereto, shall be prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2 Procedures and Programs which affect nuclear safety (but are not required by Specification 6.8) and changes thereto, shall be prepared and reviewed in accordance with the Quality Assurance Program covering the company organization or contractor organization cognizant of the work conducted under the procedure.

6.5.2.3 Proposed changes to the Appendix "A" Technical Specifications shall be prepared by qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

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

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

The Technical content of specifications presently numbered 6.5.2.5 through 6.5.2.10 are not changed except for renumbering these Specifications as 6.5.2.6. through 6.5.2.11

Specification 6.5.3.4 is to be modified as follows:

6.5.3.4 The NSG shall review:

a. The safety evaluations for 1) changes to procedures required by Specification 6.8, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

Specification 6.8.2 is to be modified as follows:

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by the Station Manager; or by (1) the Manager, Operations (2) the Manager, Technical (3) the Manager, Maintenance (4) the Deputy Station Manager or (5) the Manager, Health Physics as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

Safety Analysis

This proposed change clarifies the Technical Specifications in that it allows certain outside organizations to perform certain review functions, formerly performed by Station Supervisory Personnel. This arrangement is similar to those used during the construction of the plant. All other changes are only for clarification.

Accordingly, it is concluded that, (1) Proposed Change NPF-10-11 does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis, (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change, and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

LP:3645

ATTACHMENT A

6.1 RESPONSIBILITY

6.1.1 The Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Watch Engineer (or during his absence from the Control Room Area, a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the Vice-President of Nuclear Operations shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The Unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

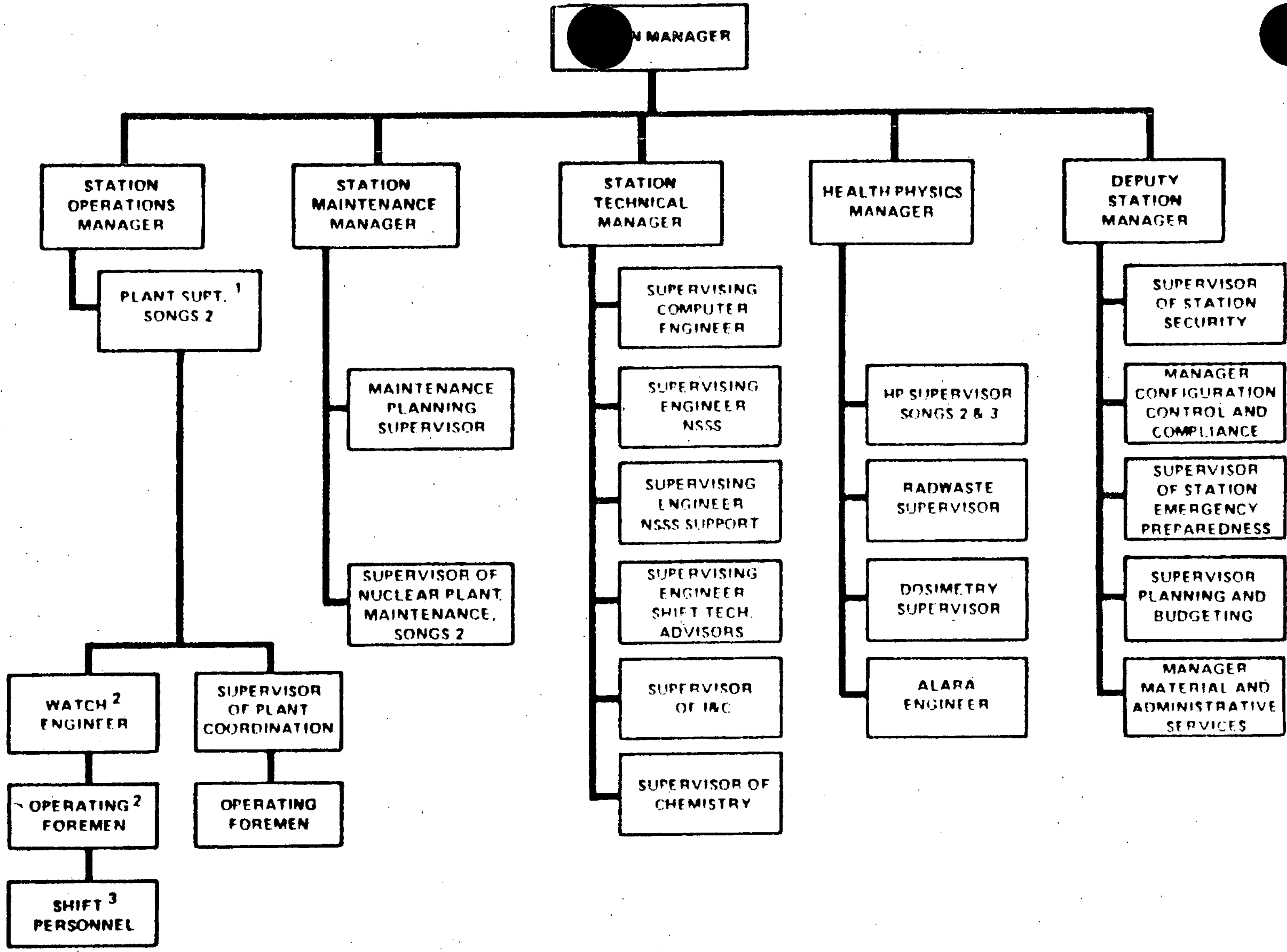
b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control Room area identified as such on Table 6.2-1.

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

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

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times.# The Fire Brigade shall not include the Watch Engineer and the 2 other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

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



1 At time of appointment to the position, Senior Reactor Operator License Required

2 Senior Reactor Operator License Required

3 Control and Assistant Control Operators are holders of Reactor Operator Licenses

Figure 6.2-2
UNIT ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION - UNIT 2

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
WE	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

WE - Watch Engineer with a Senior Reactor Operators License on Unit 2
SRO - Individual with a Senior Reactor Operators License on Unit 2
RO - Individual with a Reactor Operators License on Unit 2
AO - Auxiliary Operator
STA - Shift Technical Advisor

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

During any absence of the Watch Engineer from the Control Room Area shown in Figure 6.2-3 while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Watch Engineer from the Control Room Area shown in Figure 6.2-3 while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

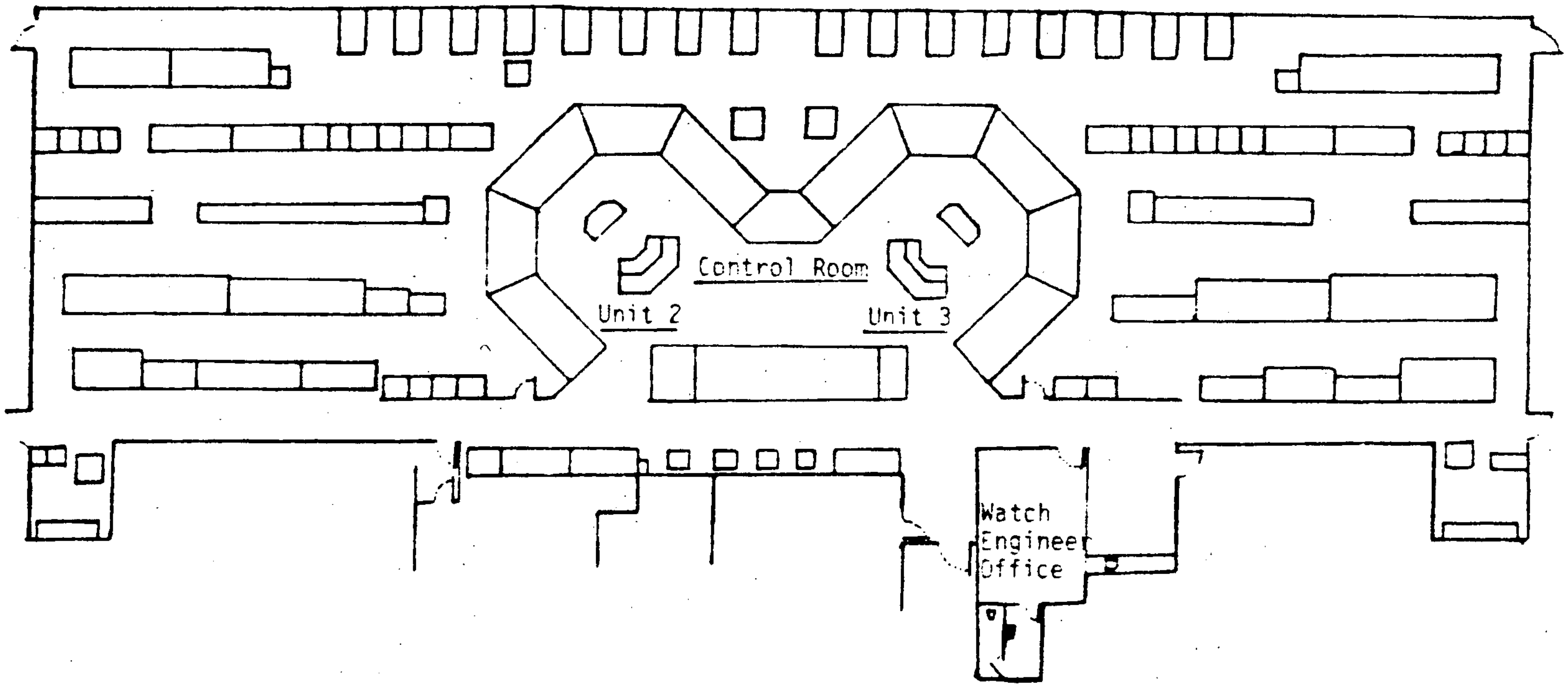


figure 6.2-3

CONTROL ROOM AREA

SAN ONOFRE NUCLEAR GENERATING STATION - UNIT 2

AUTHORITY

6.5.1.7 The Onsite Review Committee (OSRC) shall:

a. Render determinations in writing with regard to whether or not items considered under 6.5.1.6(a) above constitute unreviewed safety questions.

b. Provide written notification within 24 hours to the Manager of Nuclear Operations and NSG of disagreement between the OSRC and the Station Manager; however, the Station Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Onsite Review Committee shall maintain written minutes of each OSRC meeting that, at a minimum, document the results of all OSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Nuclear Safety Group.

6.5.2 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

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

6.5.2.2 Proposed changes to the Appendix "A" Technical Specifications shall be prepared by a qualified individual/organization. The preparation of each proposed Technical Specifications change shall be reviewed by an individual/group other than the individual/group which prepared the proposed change, but who may be from the same organization as the individual/group which prepared the proposed change. Proposed changes to the Technical Specifications shall be approved by the Station Manager.

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

ACTIVITIES (Continued)

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

6.5.2.5 Proposed tests and experiments which affect station nuclear safety and are not addressed in the FSAR or Technical Specifications shall be reviewed by the Station Manager, the Station Technical Manager, the Station Operations Manager, the Station Maintenance Manager, the Deputy Station Manager or the Health Physics Manager as previously designated by the Station Manager.

6.5.2.6 The station security program, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Manager of Nuclear Operations and to the NSG.

6.5.2.7 The station emergency plan, and implementing procedures, shall be reviewed at least once per 12 months. Recommended changes shall be approved by the Station Manager and transmitted to the Manager of Nuclear Operations and to the NSG.

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

6.5.2.9 The Station Manager shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste treatment systems.

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

6.5.3 NUCLEAR SAFETY GROUP (NSG)

FUNCTION

6.5.3.1 The Nuclear Safety Group shall function to provide independent review and audit of designated activities in the areas of:

FUNCTION (Continued)

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.3.2 NSG shall consist of a Supervisor and at least three staff specialists. The Supervisor shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of six years of professional level managerial experience in the power field. Each staff specialist shall have a Bachelor's Degree in Engineering or Physical Science and a minimum of five years of professional level experience in the field of his specialty.

The NSG shall use specialists from other technical organizations to augment its expertise in the disciplines of 6.5.2.1. Such specialists shall meet the same qualification requirements as the NSG members.

CONSULTANTS

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

REVIEW

6.5.3.4 The NSG shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

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

g. PROCESS CONTROL PROGRAM implementation.*

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15 Rev. 1, February 1979.

NOTE: Quality Assurance Program for effluent and environmental monitoring and sampling shall be in accordance with Regulatory Guide 4.15, December, 1977 prior to first exceeding 5% RATED THERMAL POWER or July 1, 1982, whichever occurs first; subsequent to this time the Quality Assurance Program shall be in accordance with Regulatory Guide 4.15, Rev. 1, February, 1979.

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

NOTE: 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 Onsite Review Committee.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed and approved by the Station Manager; or by (1) the Station Operations Manager, (2) the Station Technical Manager, (3) the Station Maintenance Manager, (4) the Station Deputy Manager or (5) the Health Physics Manager as previously designated by the Station Manager; prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 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 management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed and approved by the Station Manager; or by (1) the Station Operations Manager, (2) the Station Technical Manager, (3) the Station Maintenance Manager, (4) the Deputy Station Manager or (5) the Health Physics Manager as previously designated by the Station Manager; within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

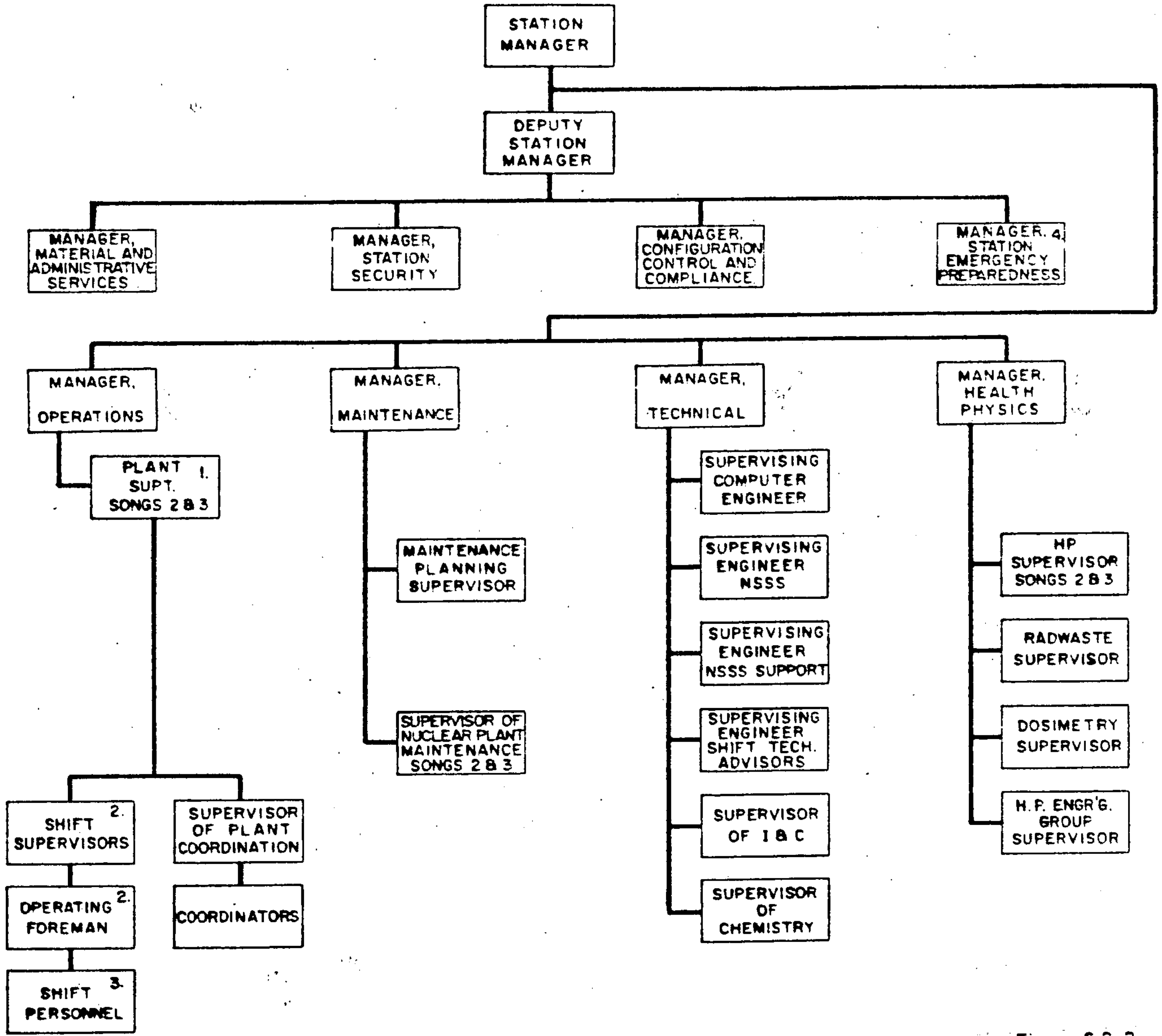
a. <u>Primary Coolant Sources Outside Containment</u>

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 high pressure safety injection recirculation, the shutdown cooling system, the reactor coolant sampling system (post-accident sampling piping only), the containment spray system, the radioactive waste gas system (post-accident sampling return piping only) and the liquid radwaste system (post-accident sampling return piping only). The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and
(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

*See Specification 6.13.1

ATTACHMENT B



1. AT TIME OF APPOINTMENT TO THE POSITION, SENIOR REACTOR OPERATOR LICENSE REQUIRED.
2. SENIOR REACTOR LICENSE REQUIRED.
3. CONTROL AND ASSISTANT CONTROL OPERATORS ARE HOLDERS OF REACTOR OPERATOR LICENSES.
4. INCLUDES FIRE PROTECTION.

Figure 6.2-2

UNIT ORGANIZATION

SAN ONOFRE NUCLEAR GENERATING STATION - UNIT 2

DESCRIPTION OF PROPOSED CHANGE NPF-10-12 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 3 OPERATING LICENSE NPF-10

This is a request to revise Appendix "A" Technical Specification 3/4.12.1 Table 3.12-1

Existing Specification:

See Attachment "A"

Proposed Specification:

Item 5 is to be deleted from page 3/4 12-5 and as a result, item g is to be deleted from page 3/4 12-6.

Reason For Proposed Change:

The San Onofre area is classified as semi-arid and local vegetation does not grow most of the year. Therefore local vegetation is an unreliable indicator of an exposure pathway. As an alternate, a statement has been added to the Section 5.0 of the Offsite Dose Calculation Manual to commit to sampling the milk of a milk producing animal found during a land use census performed in accordance with Technical Specifications 3.12.2. Sampling this milk will adequately cover this ingestion pathway.

Safety Analysis

The proposed change to the technical specifications deletes the requirement to sample through one ingestion pathway and the commitment has been made to sample through the milk ingestion pathway when it is available.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-12 does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the amendment; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

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ATTACHMENT A

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations[a]	Sampling and Collection Frequency[a]	Type and Frequency of Analyses
4. INGESTION			
a. Nonmigratory Marine Animals	3 Locations	One sample in season, or at least once per 184 days if not seasonal. One sample of each of the following species: 1. Fish-2 adult species such as perch or sheepshead. 2. Crustaceae-such as crab or lobster. 3. Mollusks-such as limpets or seahares.	Gamma isotopic analysis on edible portions.
b. Local Crops	2 Locations	Representative vegetables, normally 1 leafy and 1 fleshy collected at harvest time. At least 2 vegetables collected semiannually from each location.	Gamma isotopic analysis on edible portions semiannually and I-131 analysis for leafy crops.
5. Local Vegetation	3 Locations[g]	Monthly	Monthly gamma isotopic analysis.

TABLE 3.12-1 (Continued)

TABLE NOTATION

a. Sample locations are indicated in the ODCM

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

c. 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 which provide valid background data may be substituted.

d. 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.

e. Regulatory Guide 4.13 provides minimum acceptable performance criteria 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 purposed 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.

f. Composite samples should be collected with equipment (or equivalent) which is capable of collecting an aliquot at time intervals which are very short (e.g., hourly) relative to the compositing period (e.g., monthly).

g. 2 samples should be from the nearest offsite locations of highest calculated annual average ground-level D/Q. The third sample should be of similar vegetation characteristics and grown 15-30 km distant in the least prevalent wind direction.

DESCRIPTION OF PROPOSED CHANGE NPF-10-13 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 3 OPERATING LICENSE NPF-10

This is a request to revise Appendix "A" Technical Specification 4.6.1.3.a.

CONTAINMENT SYSTEMS, CONTAINMENT AIR LOCKS

Existing Specification

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 no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to 10 psig for at least 15 minutes.

Proposed Specification

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 is less than or equal to .01 La when determined by flow measurement, with the volume between the door seals pressurized to 9.5 ± 0.5 psig for at least 15 minutes

Reason For Proposed Change

The present specification calls for "no detectable seal leakage." This specification is difficult to verify with a pressure drop test due to the small volume of air between the pressure seals and the suceptibility of this volume to pressure change in response to temperature change. Industry experience has shown that determination of seal leakage with a flow test is a more practical method of testing. The leakage value of .01 La is also consistent with standard industry practice.

The door seal manufacturers recommend that the pressure used between the seals for leak testing not exceed 10 psig. We are therefore requesting to lower the test pressure to 9.5 ± 0.5 psig.

Safety Analysis:

This proposed change facilitates leakage testing of the airlock door seals but does not alter the intent of the technical specifications.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-13 does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the proposed change; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

DESCRIPTION OF PROPOSED CHANGE NPF-10-14 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 3 OPERATING LICENSE NPF-10

This is a request to revise Section 2.C.(5)c of Facility Operating License No. NPF-10.

Section 2.C.(5) Environmental Qualification (Section 3.11 SER, SSER#3, SSER#4)

Existing Condition

"c. By June 30, 1982, or prior to exceeding five (5) percent power, whichever comes first, SCE shall provide affirmaton of implementation of the surveillance and maintenance program procedures."

Proposed Change

c. By June 30, 1982, or prior to exceeding five (5) percent power, whichever comes first, the applicants shall provide affirmation of implementation of the maintenance program procedures.

d. Prior to startup following the first refueling outage, the applicants shall provide affirmation of implementation of the surveillance program procedures.

Reason For Proposed Change:

Provisions of proposed rule 10 CFR 50.49 state that the holder of an operating license must submit a schedule for testing or replacement of electrical equipment. The schedule must also establish a goal of final environmental qualification by the end of the second refueling outage after March 31, 1982.

The proposed rule 10 CFR 50.49 will codify the Commissions's current requirements for environmental qualification of electrical equipment and clarify what surveillance requirements are needed.

The applicants are seeking to defer the implementation date for the surveillance program portion of Section 2.C.(5)c for the following reasons:

1. The applicants are in the process of evaluating the effort required to develop and implement a surveillance and trending program for environmental qualification. Because the nuclear industry is in the early stages of developing an environmental qualification program consistent with NUREG-0588, there is very little information available to support development of a surveillance program. Accordingly, the applicants are in the process of hiring a consultant experienced in environmental qualification to assist in development of this program.

The applicants anticipate the present required Technical Specification surveillance program will form a large part of the environmental surveillance program. Therefore, the existing surveillance program is adequate until more complete guidance is available as it establishes acceptance criteria for performance of the most important systems and subsystems in the plant. The records of current surveillance tests will be used when an environmental qualification surveillance program is implemented so data can be backfitted to the beginning of cycle 1.

2. Records of environmentally qualified electrical equipment will be updated and maintained to document when equipment is replaced or further tested and qualified,

3. Environmentally qualified equipment has an installed useful life of from 2 to 40 years. Equipment qualified for less than the full life of the plant will be replaced prior to the end of qualified life.

The applicants will be in compliance with NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment" Revision 1 by June 30, 1982 for safety-related electrical equipment exposed to a harsh environment. Complete and auditable records will be available and maintained at a central location which describe the environmental qualification method.

Since the applicants will meet the requirements of NUREG-0588, Revision 1 and implement maintenance program procedures by June 30, 1982, the extension should be granted.

Safety Analysis

The proposed change delays the implementation date of the surveillance program portion of Section 2.C.(5)c. Accordingly, it is concluded that: (1) Proposed Change NPF-10-14 does not involve an unreviewed safety question as defined in 10 CFR 50.59, nor does it present significant hazard considerations not described or implicit in the Final Safety Analysis; (2) there is reasonable assurance that the health and safety of the public will not be endangered by the amendment; and (3) this action will not result in a condition which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

GVN:4108
5/5/82