ATTACHMENT A

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.

ATTACHMENT B

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 a. At least one of the following loop(s)/trains listed below shall be OPERABLE and in operation*:

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. One additional Reactor Coolant Loop/shutdown cooling train shall be OPERABLE, or

c. 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, immediatel;y initiate corrective action to return the required trains/loops to OPERABLE status or restore the required level as soon as possible.

b. With no loop/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 loop/ train to operation.

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* 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

SURVEILLANCE REQUIREMENTS

4.4.1.4.1 The required Reactor Cooling 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.

4.4.1.4.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$10% (wide range) at least once per 12 hours.

4.4.1.4.3 At least one Reactor Coolant loop or shutdown cooling train shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

DESCRIPTION OF PROPOSED CHANGE NPF-10-16 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 11 OPERATING LICENSE NPF-10

This is a request to revise Appendix "A" Technical Specification 4.6.2.1.b.4.

CONTAINMENT SPRAY SYSTEM

Existing Specification

"4. Verifying that each containment spray header riser is filled with water up to the riser vent (and overflow) valve."

Proposed Specification

4. Verifying that each containment spray header riser is filled with water to within 10 feet of the lowest spray ring.

Reason for Proposed Change

The original method of filling the containment spray riser was to close the spectacle blind at the top of the riser and open the adjacent vent valves and fill. The riser was filled with demineralized water through the air test connection with a temporary hose. The riser was determined to be full when water issued from the vent valve.

This method of filling required sending personnel up to the 143 foot level of the containment structure and was predicated upon there being an auxiliary lift on the polar crane to allow for easy access. Since this auxiliary lift is no longer part of the design, a costly temporary scaffold is now required to access the top of the riser.

The proposed specification would not require personnel access to the upper containment dome area. The new procedure utilizes a precision pressure gauge at the bottom of the riser to determine and monitor the water level in the spray header riser during filling. The spectacle blind is left open and the vent valve closed during the operation. The system is vented through the spray nozzles and the riser is filled to within 10 feet of the lowest spray ring.

Safety Analysis

The fill elevation under the current specification is the vent valve location. The proposed specification fills the riser to within 10 feet of the lowest spray ring which is 8 feet 3 inches below the vent valve location. The slightly reduced water level in the riser will not affect the results of the safety analysis which used a fill level of 10 feet below the lowest spray ring.

Accordingly, it is concluded that: (1) Proposed change NPF-10-16 does not 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-10-21 and Safety Analysis
Amendment Application No. 11 to Operating License NPF-10

This is a request to revise Appendix "A" Technical Specification 6.9.1.13.b by the addition of a note consistant with USNRC Regulatory Guide 1.16, Rev. 4 and proposed rule change to 10 CFR 50.73 (Ref. Fed. Reg. Vol. 47, No. 88 dated May 6, 1982).

Existing Specification

See Attachment "A"

Proposed Specification

Add a note following Section 6.9.1.13.b as follows:

Note: Those cases where a system or component is removed from service as part of a planned evolution, in accordance with an approved procedure, and returned to service within the limits of the applicable Limiting Condition for Operation Action Statement, need not be reported under this criterion.

Safety Analysis

This change only affects the reporting requirements of Technical Specification Administrative Controls and is consistant with USNRC Regulatory Guide 1.16-Rev. 4 - Reporting of Operating Information - Appendix A Technical Specifications as well as a proposed rule change to 10 CFR 50.73 published in the Federal Register, Volume 47, No. 88, dated May 6, 1982.

Accordingly, it is concluded that: (1) This change does not 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.

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ATTACHMENT "A"

THIRTY DAY WRITTEN REPORTS

6.9.1.13 The types of events listed below shall be the subject of written reports to the NRC Regional Administrator within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.12.c above designed to contain radioactive material resulting from the fission process.

HAZARDOUS CARGO TRAFFIC REPORT

6.9.1.14 Hazardous cargo traffic on Interstate 5 (I-5) and the AT&SF railway shall be monitored and the results submitted to the NRC Regional Administrator once every three years.

SPECIAL REPORTS

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

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.

DESCRIPTION OF PROPOSED CHANGE NPF-10-23 AND SAFETY ANALYSIS AMENDMENT APPLICATION NO. 11 TO OPERATING LICENSE NPF-10

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

Existing Specifications

See Attachment "A"

Proposed Specifications

The proposed specifications are as follows:

Page: 3/4 3-15 — Technical Specification 3.3.2, Table 3.3-3, Item 6.b Containment Cooling (CCAS) Manual SIAS is required to be operable in Modes 1, 2 and 3, but Mode 4 should be added under applicable Modes. Mode 4 is a requirement in Manual SIAS Items 1.a, 3.b, and 9.b of Table 3.3-3 and 6.b of Table 4.3-2.

Page: 3/4 3-26 — Technical Specification 3.3.2, Table 3.3-4 notation (1) should state "bypass shall be automatically removed whenever 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.

Page: 3/4 3-31 — Technical Specification 3.3.2, Table 4.3-2 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 from Item 2 Table 4.3-2. Item 5, Recirculation (RAS), 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 4.3-2.

Page: 3/4 8-18 through 3/4 8-25 — Technical Specification 3.8.4.1, Table 3.8-1 has various typographical errors and are to be changed as follows:

<u>Page 3/4 8-18</u>

Line 1 — should read "Containment Normal Cooling Fan E-397" not "E-387."

Line 6 — Should read "Hydrogen Recombiner E-145 Power Panel L-180" not "L-160."

Line 11 — Should read "Hydrogen Recombiner E-146 Power Panel L-181" not "L-161."

Page 3/4 8-19

Line 11 Should read "Cont. Cooling Unit E-394 Circ. Water Outlet HV-9940EB" not "HV-9930EB."

Line 25 Should read "Reactor Cavity Cooling Fan A-319" not "A-313."

Page 3/4 8-20

Line 10 The "(A)" should be deleted.

Line 11 Should read "Containment Elevator P-003" not "P-002(A)."

Line 15 Primary Device Number is to be "2BE10" not "2BE11."

Page 3/4 8-21

Line 17 Should read "Containment Normal Cooling Supply Isol. Valve HV-9900" not "HV-9400."

Line 19 Should read "Movable Incore Detector Drive Pack W-338B" not "W-3383."

Page 3/4 8-22

Line 21 Should read "Lower Level Air Circulator A-034" not "A-024."

Line 22 Should read "Cont. Cooling Unit E-396" not "E-346."

Line 24 Should read "Cont. Cooling Unit E-398" not "E-348."

Page 3/4 8-23

Line 10 Should read "Containment Normal Cooling Fan E-396" not "E-398."

Page 3/4 8-24

Line 5 Should read "Backup Pressurizer Heater E-620" not "E-602."

Line 24 Should read "Containment Reactor Cavity Cooling Fan A-319" not "E-319."

Page 3/4 8-25

Line 17 Should read "Panel 2LP16" not "3LP16."

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-10-23 are editorial or typographical and do not change the intent of the Technical Specifications.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-23 does not 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 conditon 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 3.3-3 (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 3.3-4 (Continued)

TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* 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 is greater than or equal to 500 psia.

(2) 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.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value 3165V, trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages.

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

*Variable setpoints are for use only during <u>normal</u>, <u>controlled</u> plant heatups and cooldowns.

**Above normal background.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2B0106	2BLP0101	Containment Normal Cooling Fan E-387
2B0107	2BLP0102	CEDM Cooling Supply Fan E-403B
2B0109	2BLP0103	CEDM Cooling Supply Fan E-403A
2B0111	2BLP0104	Standby Containment Normal Cooling Fan E-393
2B0209	2BLP0201	Containment Normal Cooling Fan E-394
2B0406	2BLP0301	Hydrogen Recombiner E-145 Power Panel L-160
2B0409	2BLP0302	Upper Dome Air Circulator A-071
2B0410	2BLP0303	Containment Emergency Fan E-399
2B0411	2BLP0304	Containment Emergency Fan E-401
2B0419	2BLP0305	Standby Upper Dome Air Circulator A-074
2B0606	2BLP0401	Hydrogen Recombiner E-146 Power Panel L-161
2B0609	2BLP0402	Upper Dome Air Circulator A-072
2B0610	2BLP0403	Containment Emergency Fan E-400
2B0611	2BLP0404	Containment Emergency Fan E-402
2B0619	2BLP0405	Standby Upper Dome Air Circulator A-073
2B0809	2BLP0501	Containment Normal Cooling Fan E-396
2B0811	2BLP0601	Containment Normal Cooling Fan E-398
2B0903	2BLP0701	Containment Recirculation Unit E-333
2B0906	2BLP0702	Polar Crane (Containment) R001 (C)
2B0907	2BLP0703	Standby Control Element Drive Mechanism Cooling Supply Fan E-404A
2B0909	2BLP0704	Standby CEDM Cooling Supply Fan E-404B
2B0911	2BLP0705	Containment Recirculating Unit Heater E-568
2BA02	2BLP0812	CCW from RCP P-001 Seal Heat Exchanger TV-9144
2BA03	2BLP0813	CCW from RCP P-003 Seal Heat Exchanger TV-9154
2BA04 (2BA04-A)	2BLP0801	CEDM Cooling Supply Fan E-403A (Enclosure Heater)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BA04 (2BA04-B)	2BLP0802	CEDM Cooling Supply Fan E-403B (Enclosure Heater)
2BA04 (2BA04-C)	2BLP0814	Standby Containment Normal Cooling Fan E-393 (Enclosure Heater)
2BA04 (2BA04-D)	2BLP0826	Containmnt Normal Cooling Fan E-394 (Enclosure Heater)
2BA04 (2BA04-E)	2BLP0828	Containment Normal Cooling Fan E-397
2BA08	2BLP0803	Movable Incore Detector Drive Package W338A
2BA11	2BLP0905	Cont. Structure Electric Heater E-467
2BA25	2BLP0910	Cont. Cooling Unit E-393 Circ. Water Outlet HV-9940FB
2BA26	2BLP0911	Cont. Cooling Unit E-394 Circ. Water Outlet HV-9930EB
2BA27	2BLP0912	Cont. Cooling Unit E-397 Circ. Water Outlet HV-9940DB
2BA31	2BLP0913	Cont. Cooling Unit E-393 Circ. Water Outlet HV-9940FC
2BA32	2BLP0914	Cont. Cooling Unit E-394 Circ. Water Inlet HV-9940EC
2BA33	2BLP0915	Cont. Cooling Unit E-397 Circ. Water Inlet HV-9940DC
2BA36	2BLP0808	RCP 1A Oil Lift Pump 1A1 P-260
2BA37	2BLP0809	RCP 1B Oil Lift Pump 1B1 P-264
2BA38	2BLP0810	RCP 2B Oil Lift Pump 2B1 P-262
2BA39	2BLP0901	Reactor Coolant Drain Pump (W) P-023
2BA40	2BLP0811	RCP 2A Oil Lift Pump 2A1 P-266
2BA41	2BLP0817	RCP 1A Anti Rev. Rotation Device Lube Pump 1 P-399
2BA42	2BLP0818	RCP 2B Anti Rev. Rotation Device Lube Pump 1 P-401
2BA43	2BLP0819	RCP 1B Anti Rev. Rotation Device Lube Pump 1 P-403
2BA44	2BLP0820	RCP 2A Anti Rev. Rotation Device Lube Pump 1 P-405
2BA45	2BLP0902	Reactor Cavity Cooling Fan A-313
2BA46	2BLP0903	Standby Reactor Cavity Cooling Fan A-321

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TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BA47	2BLP0807	Charging Line to Reactor Cooling Loop 1A HV-9203
2BA49	2BLP0821	Reactor Cavity Cooling Unit C HV-9905C
2BA50	2BLP0822	Reactor Cavity Cooling Unit A HV-9905A
2BA51	2BLP0804	Quench Tank to Reactor Drain Tank HV-9101
2BA55	2BLP0805	RCP Bleed Off to Quench Tank HV-9216
2BA57	2BLP0916	CEDM Cooling Unit E-403 CCW Outlet HV-9907AA
2BA58	2BLP0917	CEDM Cooling Unit E-403 CCW Inlet HV-9907AC
2BA59	2BLP0806	Safety Injection Tank to Reactor Drain Tank HV-9335
2BA60	2BLP0904	Welding Receptacles Containment (50 KVA)
2BA62	2BLP0824	Recept. for Portable Cont. Sump Pump (H.P.) P-005 (A)
2BA63	2BLP0906	Containment Elevator P-002 (A)
2BA65	2BLP0815	Lower Level Air Circulator A-031
2BA66	2BLP0816	Lower Level Air Circulator A-033
2BE09	2BLP1001	Saf. Inj. Tank Drain to Refueling Wtr Tank HV-9334
2BE11	2BLP1002	Saf. Inj. Tk T-007 to Reactor Coolant Loop 1B HV-9350
2BE11	2BLP1003	Saf. Inj. Tk T-009 to Reactor Coolant Loop 2A HV-9360
2BE17	2BLP1010	Auxiliary Spray to Pressurizer HV-9201
2BE21	2BLP1012	CCW Noncritical Cont. Inlet Isolation Valve HV-6223
2BE25	2BLP1005	Shutdn Coolant Flow from Reac. Coolant Loop 2 HV-9337
2BE26	2BLP1015	Reac. Coolant Drain Tk Sample Cont. Isolation HV-0516
2BE27	2BLP1016	Containment Isolation Reactor Coolant Drain to Radwaste System HV-7512
2BE30	2BLP1017	Quench Tank Vapor Sample Cont. Isol. HV-0514
2BE31	2BLP1004	Containment Sump to Radwaste Sump HV-5803
2BE33	2BLP1021	Containment Purge Inlet HV-9949
2BE35	2BLP1018	Containment Emergency Sump Outlet HV-9305

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BE46	2BLP1011	CCW Noncritical Containment Isolation Valve HV-6336
2BF08	2BLP0823	Containment Sump Pump P-008
2BF09	2BLP1220	Containment Sump Pump P-007
2BJ05	2BLP1101	Shutdn Coolant Flow from Reac. Coolant Loop 2 HV-9339
2BJ06	2BLP1104	Saf. Inj. Tk T-008 to Reactor Coolant Loop 1A HV-9340
2BJ07	2BLP1105	Saf. Inj. Tk T-010 to Reactor Coolant Loop 2B HV-9370
2BJ17	2BLP1123	RCP Bleed off to Volume Control Tank HV-9217
2BJ21	2BLP1106	Cont. Isol. Safety Injection Tank Vent Header HV-7258
2BJ22	2BLP1115	Reactor Coolant Hot Leg Sample Cont. Isol. HV-0508
2BJ23	2BLP1116	Reactor Coolant Hot Leg Sample Cont. Isol. HV-0517
2BJ26	2BLP1117	Pressurizer Vapor Sample Containment Isol. HV-0510
2BJ27	2BLP1121	Pressur. Surge Line Liquid Smpl. Cont. Isol. HV-0512
2BJ29	2BLP1110	Containment Purge Outlet HV-9950
2BJ30	2BLP1102	Hydrogen Purge Exhaust Unit Inlet HV-9917
2BJ31	2BLP1103	Hydrogen Purge Supply Unit Discharge HV-9946
2BJ34	2BLP1118	Containment Emergency Sump Outlet HV-9304
2BJ47	2BLP1124	Containment Normal Cooling Supply Isol. Valve HV-9400
2BJ48	2BLP1125	Containment Normal Cooling Return Isol. Valve HV-9971
2BN04	2BLP1201	Movable Incore Detector Drive Pack W-3383
2BN07	2BLP1304	Containment Structure Electric Heater E-466
2BN21	2BLP1206	Charging Line to Reactor Coolant Loop 2A HV-9202
2BN24	2BLP1301	Reactor Cavity Cooling Fan A-320
2BN25	2BLP1302	Standby Reactor Cavity Cooling Fan A-322
2BN26	2BLP1226	CCW from RCP P-004 Seal Heat Exchanger TV-9164
2BN27	2BLP1227	CCW from RCP P-002 Seal Heat Exchanger TV-9174

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BN28	2BLP1207	Reactor Cavity Cooling Unit D HV-9905D
2BN29	2BLP1208	Reactor Cavity Cooling Unit B HV-9905B
2BN30	2BLP1209	RCP 1A Oil Lift Pump 1A2 P-261
2BN31	2BLP1210	RCP 1B Oil Lift Pump 1B2 P-265
2BN32	2BLP1211	RCP 2B Oil Lift Pump 2B2-263
2BN33	2BLP1212	RCP 2A Oil Lift Pump 2A2-267
2BN34	2BLP1303	Reactor Coolant Drain Tank Pump (E) P-022
2BN37	2BLP1213	RCP 1A Anti Rev. Rotation Device Lube Pump 2 P-400
2BN38	2BLP1214	RCP 2B Anti Rev. Rotation Device Lube Pump 2 P-402
2BN39	2BLP1215	RCP 1B Anti Rev. Rotation Device Lube Pump 2 P-404
2BN40	2BLP1216	RCP 2A Anti Rev. Rotation Device Lube Pump 2 P-406
2BN42	2BLP1305	Welding Recpt. Cont. (50KVA) 2R005A, 2R005b, 2R005C
2BN43	2BLP1217	CEA Change Mechanism Transfer Machine Control Console (8 KVA) L-023
2BN44	2BLP1306	Welding Recpt. Cont. (50 KVA) 2R007A, 2R007B, 2R007C
2BN45	2BLP1218	Refueling Pool End Junction Box (8KVA) L-371
2BN46	2BLP1308	Welding Recpt. Cont. (50KVA) 2R013A, 2R013B, 2R013C
2BN47	2BLP1219	Receptable for Portable Cont. Sump Pump (1hp) P-005
2BN49	2BLP1319	Equipment Hatch 200R, Electrical Hoist Z-028, Z-029
2BN52	2BLP1221	Lower Level Air Circulator A-032
2BN53	2BLP1222	Lower Level Air Circulator A-024
2BN56	2BLP1310	Cont. Cooling Unit E-346 Circ. Water Outlet HV-9940BB
2BN57	2BLP1311	Cont. Cooling Unit E-396 Circ. Water Inlet HV-9940BC
2BN58	2BLP1312	Cont. Cooling Unit E-348 Circ. Water Outlet HV-9940CB
2BN59	2BLP1313	Cont. Cooling Unit E-398 Circ. Water Inlet HV-9940CC
2BN60	2BLP1314	CEDM Cooling Unit E-404 CCW Outlet HV-9907BA

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BN61	2BLP1315	CEDM Cooling Unit E-404 CCW Inlet HV-9907BC
2BN62 (2BN62-A)	2BLP1223	Containment Recirculation Unit A-353 (Motor Enclosure Heater)
2BN62 (2BN62-B)	2BLP1224	CEDM Cooling Supply Fan E-404A (Motor Enclosure Heater)
2BN62 (2BN62-C)	2BLP1225	CEDM Cooling Supply Fan E-404B (Motor Enclosure Heater)
2BN62 (2BN62-H)	2BLP1202	Containment Normal Cooling Fan A-398 (Motor Enclosure Heater)
2BN62 (2BN62-G)	2BLP1228	Containment Normal Cooling Fan E-398 (Motor Enclosure Heater)
L0108	L0101	Panel 2LP4 Emergency Lighting
L0118	L0101	Panel 2LP11 Emergency Lighting
L0120	L0101	Panel 2LP16 Emergency Lighting
2BHP0201	2B0205	Backup Pressurizer Heater E-607
2BHP0202	2B0205	Backup Pressurizer Heater E-608
2BHP0203	2B0205	Backup Pressurizer Heater E-609
2BHP0204	2B0205	Backup Pressurizer Heater E-610
2BHP0301	2B0206	Backup Pressurizer Heater E-611
2BHP0302	2B0206	Backup Pressurizer Heater E-612
2BHP0303	2B0206	Backup Pressurizer Heater E-613
2BHP0304	2B0206	Backup Pressurizer Heater E-614
2BHP0101	2B0210	Proportional Pressurizer Heater E-601
2BHP0102	2B0210	Proportional Pressurizer Heater E-602
2BHP0103	2B0210	Proportional Pressurizer Heater E-603
2BHP0401	2B0402	Backup Pressurizer Heater E-615

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2BHP0402	2B0402	Backup Pressurizer Heater E-616
2BHP0403	2B0402	Backup Pressurizer Heater E-617
2BHP0404	2B0402	Backup Pressurizer Heater E-618
2BHP0601	2B0805	Backup Pressurizer Heater E-619
2BHP0602	2B0805	Backup Pressurizer Heater E-602
2BHP0603	2B0805	Backup Pressurizer Heater E-621
2BHP0604	2B0805	Backup Pressurizer Heater E-622
2BHP0701	2B0806	Backup Pressurizer Heater E-623
2BHP0702	2B0806	Backup Pressurizer Heater E-624
2BHP0703	2B0806	Backup Pressurizer Heater E-625
2BHP0704	2B0806	Backup Pressurizer Heater E-626
2BHP0501	2B0810	Proportional Pressurizer Heater E-604
2BHP0502	2B0810	Proportional Pressurizer Heater E-605
2BHP0503	2B0810	Proportional Pressurizer Heater E-606
2BHP0801	2B0602	Backup Pressurizer Heater E-627
2BHP0802	2B0602	Backup Pressurizer Heater E-628
2BHP0803	2B0602	Backup Pressurizer Heater E-639
2BHP0804	2B0602	Backup Pressurizer Heater E-630
2BY40	2BLP1013	Cont. Bldg. Emer. A/C Unit E-399 (Motor Enclos. Htr.)
2BY40	2BLP1014	Cont. Bldg. Emer. A/C Unit E-401 (Motor Enclos. Htr.)
2BZ32	2BLP1111	Reactor Coolant Regen. Heat Exch. Isol. Valve TV-9267
2BZ38	2BLP1112	Containment Bldg. Emergency A/C Unit E-400
2BZ38	2BLP1126	Containment Bldg. Emergency A/C Unit E-403
2Q01704	2Q017 (Main Breaker)	Containment Reactor Cavity Cooling Fan E-319 (Motor Enclosure Heater)
2Q01706	2Q017 (Main Breaker)	Containment Reactor Cavity Cooling Fan A-321 (Motor Enclosure Heater)

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

Primary Device Number	Backup Device Number	Service Description
2Q01724	2Q017 (Main Breaker)	Containment Sump Inlet Flow 2FT5799A/B, 2FT5802A/B
2Q02801	2Q028 (Main Breaker)	RCP P-001 (Motor Enclosure Heater)
2Q02802	2Q028 (Main Breaker)	RCP P-004 (Motor Enclosure Heater)
2Q02803	2Q028 (Main Breaker)	RCP P-002 (Motor Enclosure Heater)
2Q02804	2Q028 (Main Breaker)	Containment Reactor Cavity Cooling Fan A-320 (Motor Enclosure Heater)
2Q02805	2Q028 (Main Breaker)	RCP P-003 (Motor Enclosure Heater)
2Q02808	2Q028 (Main Breaker)	Containment Reactor Cavity Cooling Fan (Motor Enclosure Heater)
2Q03904	2Q039 (Main Breaker)	Dome Circulating Fan A-071 (Motor Enclosure Heater)
2Q03906	2Q039 (Main Breaker)	Dome Circulating Fan A-074 (Motor Enclosure Heater)
2Q04104	2Q041 (Main Breaker)	Standby Dome Circulating Fan A-072 (Motor Enclosure Heater)
2Q04106	2Q041 (Main Breaker)	Standby Dome Circulating Fan A-073
2D5P108	2D503	Panel 2LP4 Emergency Lighting
2D5P109	2D503	Panel 2LP11 Emergency Lighting
2D5P118	2D503	Panel 3LP16 Emergency Lighting
2A0101	2A0102	Reactor Coolant Pump P-001
	2A0104	Reactor Coolant Pump P-001
	2A0105	Reactor Coolant Pump P-001

DESCRIPTION OF PROPOSED CHANGE NPF-10-27 AND SAFETY ANALYSIS AMENDMENT APPLICATION NO. 11 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:

Page 3/4 3-10, 3/4 3-11, and 3/4 3-12	Technical Specification 3.3-1, Table 4.3-1, refers to table notation (10) under channel calibration for Functional Units 2, 3, 9, 10 and 14. Table Notation (10) makes reference to "Detector Plateau Curves" which exist only for source range (BF_3) detectors. Source range detectors are not associatd with any of the functional units referring to this table notation nor do source range detectors provide any automatic reactor protection functions. Therefore, Table Notation (10) should be deleted from page 3/4 3-12 and reference to Table Notation (10) for Functional Units 2, 3, 9, 10 and 14 should be deleted from pages 3/4 3-10 and 3/4 3-11.
Page 3/4 5-3	Technical Specification 3.5.2.d, a Recirculation Actuation Signal is received not a "Sump" Recirculation Actuation Signal. Therefore, "Sump" should be deleted.
Page 3/4 5-4	Technical Specification 4.5.2.d.1 should read "when RCS pressure is simulated greater than or equal to 700 psig", not "700 psia" to be consistent with SONGS 2&3 FSAR Section 5.2.2.11.2.2, page 5.2-7B.
Page 3/4 6-6	Technical Specification 4.6.1.3.a should read "door seals pressurized to 9.5 + 0.5 psig for at least 15 minutes." The "greater than or equal to" was left in the Technical Specification inadvertently.
Page 3/4 6-14	Technical Specification 4.6.2.1.a should read as follows: 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 the correct position with suction aligned to the RWST.

Page
3/4 6-22 Technical Specification Table 3.6-1, Penetrations Number 16A and 16B, valves HV-0500, 0501, 0502 and 0503 should have an asterisk (*) to denote that these valves may be opened on an intermittent basis under administrative control in order to be able to satisfy surveillance requirement 4.6.4.1.

Page

3/4 11-9 Technical Specification Table 4.11-2 should indicate the superscript "g" on all "Prinicipal Gamma Emitters" in the activity analysis column. This superscript was inadvertently ommitted for both 42 inch and 8 inch containment purge release types.

Page
B 3/4 4-1 Bases for Technical Specification 3/4 4-1 should read "The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR greater than or equal to 1.20... " not " ... above 1.19." This change will be consistent with Technical Specification 2.1.1.1.

Page
6-2 Figure 6.2-1, Offsite Organization. The word "Construction" contained in "Vice President (Engineering and Construction" is misspelled as "Contruction."

Page
6-25 Technical Specification 6.15.1.1.a, "determination" is misspelled as "detemrination."

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-10-27 are editorial or typographical and do not change the intent of the Technical Specifications.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-27 does not 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 conditon which significantly alters the impact of the station on the environment as described in the NRC Final Environmental Statement.

HPemberton:5422

ATTACHMENT A

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4), R(10)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)(10)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5,10)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5,10)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),S(7) R(4,5,10),M(8)	M(11),R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2
16. Reactor Coolant Flow-Low	S	R	M	1, 2
@17. Seismic-High	S	R	M	1, 2
18. Loss of Load	S	N.A.	M	1 (9)

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TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

(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 signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations 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 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 (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) 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 RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - Above 55% of RATED THERMAL POWER.

(10) - Detector plateau curves shall be obtained, evaluated, and compared to manufacturer's data.

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

(12) - 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.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. One OPERABLE charging pump capable of taking suction from either the boric acid makeup tank or the refueling water storage tank.

d. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem 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. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 400 psia.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

	Valve Number	Valve Function	Valve Position
a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	FV0306	SDC Bypass Flow Control	LOCKED OPEN (THROTTLED)(MANUAL)
h.	14-153		LOCKED CLOSED (MANUAL)
i.	14-081		LOCKED OPEN (MANUAL)
j.	14-082		LOCKED OPEN (MANUAL)
k.	HV9420	Hot Leg Injection Isolation	CLOSED
l.	HV9434	Hot Leg Injection Isolation	CLOSED
m.	HV9316	SDC(HX) Flow Control	OPEN (THROTTLED)(AIR REMOVED)
n.	10-068	RWST Isolation	LOCKED OPEN (MANUAL)
o.	14-78	HV9316 Isolation	LOCKED OPEN (MANUAL)
p.	14-80	HV9316 Isolation	LOCKED OPEN (MANUAL)

b. At least once per 31 days by:

1. Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2. 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 the correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2. Of the areas affected within containment at the completion of containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when RCS pressure is simulated greater than or equal to 700 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when simulated RCS pressure is greater than or equal to 376 psia.

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 that seal leakage is less than or equal to 0.01 La when determined by flow measurement, with the volume between the door seals pressurized to greater than or equal to 9.5 ± 0.5 psig for at least 15 minutes,

b. By conducting overall air lock leakage tests at not less than P_a (55.7 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.

*Exemption to Appendix J of 10 CFR 50.

CONTAINMENT SYSTEMS

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 RWST 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 and 3.

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 HOT SHUTDOWN within the following 6 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 that is not locked, sealed, or otherwise secured in position is positioned to take suction from the RWST on a Containment Spray Actuation (CSAS) test signal.

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 on a Containment Spray Actuation 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.

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-376*	Safety injection drain to RWST	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
9	HV-9337#@	Shutdown cooling to LPSI pumps	NA
9	HV-9377#@	Shutdown cooling to LPSI pumps	NA
9	HV-9336#@	Shutdown cooling to LPSI pumps	NA
9	HV-9379#@	Shutdown cooling to LPSI pumps	NA
10A	HV-0352A#	Containment pressure detectors	NA
10B	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10B	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
16A	HV-0500	Post LOCA hydrogen monitor	NA
16A	HV-0501	Post LOCA hydrogen monitor	NA
16B	HV-0502	Post LOCA hydrogen monitor	NA
16B	HV-0503	Post LOCA hydrogen monitor	NA
20	2"-321-C-396*	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
27A	HV-0352D#	Containment pressure detectors	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA
40A	HV-0352B#	Containment pressure detectors	NA
67	HV-9434	Hot leg injection	NA
68	2"-130-C-334	Charging line to auxiliary spray	NA
70	2"-037-C-145	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-145	Auxiliary steam inlet to utility stations	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C#	Containment pressure detectors	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment Purge 42 inch	P Each Purge[b,c]	P Each Purge[b]	Principal Gamma Emitters	1×10^{-4}
			H-3	1×10^{-6}
8 inch	M[b] Grab Sample	M[b]	Principal Gamma Emmitters	1×10^{-4}
			H-3	1×10^{-6}
C. 1. Condenser Evacuation System	M[b] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
2. Plant Vent Stack	W[b,e]	W[b]	H-3	1×10^{-6}
D. All Release Types as listed in B and C above.	Continuous[f] Sampler	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f] Sampler	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f] Sampler	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f] Sampler	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f] Monitor	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.19 during all normal operations and anticipated transients. As a result, in MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour since no safety analysis has been conducted for operation with less than 4 reactor coolant pumps or less than two reactor coolant loops in operation.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops/trains (either RCS or shutdown cooling) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling trains be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in Modes 4 and 5 with one or more RCS cold legs less than or equal to 235°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown with RCS cold leg temperature greater than 235°F. In the event that no safety valves are OPERABLE and for RCS cold leg temperature less than or equal to 235°F, the operating shutdown cooling relief valve, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

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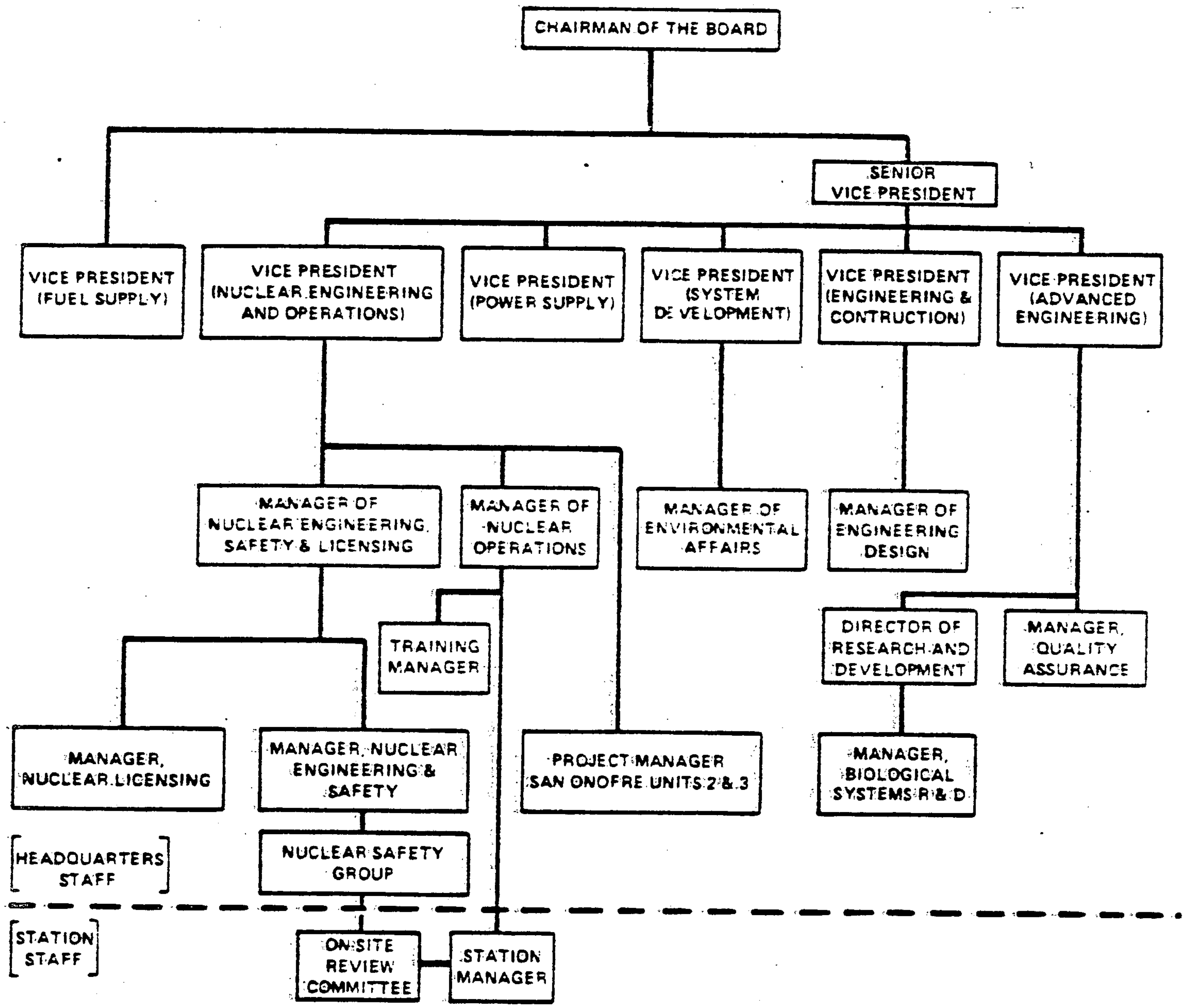

Figure 6.2-1
OFFSITE ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION – UNIT 2

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:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. 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);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the OSRC.

2. Shall become effective upon review and acceptance by the OSRC.

6.15 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and solid)

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

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the OSRC. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the detemrination that the change could be made in accordance with 10 CFR 50.59;

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

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

   d. 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;

FEB 17 [illegible]

DESCRIPTION OF PROPOSED CHANGE NPF-10-29 AND SAFETY ANALYSIS AMENDMENT APPLICATION NO. 11 OPERATING LICENSE NPF-10

This is a request to revise Appendix "A" Bases for Technical Specification 3/4.6.1.4.

INTERNAL PRESSURE

Existing Specification

See Attachment "A"

Proposed Specification

Add the following to the end of the first paragraph of Bases for Technical Specification 3/4.6.1.4 page B 3/4 6-2.

> "3) the assumptions used for the initial conditions of the LOCA safety analysis remain valid."

Reasons for Proposed Change

The containment internal pressure limit is used as an initial condition for calculating the minimum containment back pressure for ECCS performance provided in the FSAR Section 6.2.1.5. Therefore, the ECCS performance analysis provides the basis for the containment internal pressure limit specified in the Technical Specifications.

Safety Analysis

Proposed Change NPF-10-29 is an editorial change and does not change the intent of the Technical Specifications.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-29 does not 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.

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ATTACHMENT A

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 5.0 psig and 2) the containment peak pressure does not exceed the design pressure of 60 psig during LOCA or steam line break conditions.

The maximum peak pressure expected to be obtained from a LOCA or steam line break event is 55.7 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 57.2 psig which is less than the design pressure and is consistent with the accident 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 accident analysis for a steam line break accident.

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 55.7 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, the chemical and visual examination of the sheathing filler grease, and the Type A leakage tests are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Proposed Revision 3 to Regulatory Guide 1.35, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979; and Proposed Regulatory Guide 1.35.1, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," April 1979.

DESCRIPTION OF PROPOSED CHANGE NPF-10-34 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO. 11 TO OPERATING LICENSE NPF-10

This is a request to revise Table 2.2-1 of Specification 2.2.1.

This Proposed Change, (1) deletes reference to percent of full differential pressure and inserts the actual instrument calibration values, and (2) reflects a recalculation of the instrument setpoints.

Existing Specification:

Page	Functional Unit	Trip Setpoint	Allowable Value
2-3	11. Reactor Coolant Flow-Low		
	a) DN Rate	$\leq$ 0.3%/sec (6)(8)	$\leq$ 0.315%/sec (6)(8)
	b) Floor	$\geq$ 60% (6)(8)	$\geq$ 55% (6)(8)
	c) Step	$\leq$ 10% (6)(8)	$\leq$ 13% (6)(8)

Page 2-4 (6) DN RATE, % of reference value, is the maximum decrease rate of the trip setpoint.

FLOOR, % of reference value, is the minimum value of the trip setpoint.

STEP, % of reference value, is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor. The reference value is that of the input signal at operating flow and coolant temperature.

Proposed Specifications:

11. Reactor Coolant Flow-Low

Functional Unit	Trip Setpoint	Allowable Value
a. DN Rate	$\leq$ 0.22 psid/sec (6)(8)	$\leq$ 0.231 psid/sec (6)(8)
b. Floor	$\geq$ 13.2 psid (6)(8)	$\geq$ 12.1 psid (6)(8)
c. Step	$\leq$ 6.82 psid (6)(8)	$\leq$ 7.231 psid (6)(8)

To be consistent with this change, Table Notation (6) for Table 2.2.1 and is to read as follows:

(6) DN RATE, is the maximum decrease rate of the trip setpoint.

FLOOR, is the minimum value of the trip setpoint.

STEP, is the amount by which the trip setpoint is below the input signal unless limited by DN Rate or Floor.

Reason for Proposed Change:

This proposed change:

(1) Allows easier setpoint verification as the instruments are set to the values actually shown in the Technical Specifications.

(2) Modifies the setpoint values as the result of calculations performed during the startup program.

Safety Analysis:

These changes reflect a change from preliminary data to final data. The calculations have been verified in accordance with the procedures of the qualified and approved Q.A. Program.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-34 does not 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.

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