ATTACHMENT A

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (Note 4a)	21.2
(c) CCW Valves (Note 4b)	23.2*
(d) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (Note 4a)	21.0
(c) CCW Valves (Note 4b)	23.0*
(d) Emergency Cooling Fans	21.0*
b. CIAS	
Containment Isolation	10.9* (NOTE 2)
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

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INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
5. <u>Steam Generator Pressure - Low</u>	
a. MSIS	
(1) Main Steam Isolation (MSIV)	20.9
(2) Main Feedwater Isolation	10.9
6. <u>Refueling Water Storage Tank - Low</u>	
a. RAS	
(1) Containment Sump Valves Open	50.7*
(2) ECCS Miniflow Valves Shut	40.7*
7. <u>4.16 kv Emergency Bus Undervoltage</u>	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. <u>Steam Generator Level - Low (and No Pressure-Low Trip)</u>	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (steam/DC train)	30.9 (Note 6)
9. <u>Steam Generator Level - Low (and ΔP - High)</u>	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (Steam/DC train)	30.9 (Note 6)
10. <u>Control Room Ventilation Airborne Radiation</u>	
a. CRIS	
(1) Control Room Ventilation - Emergency Mode	Not Applicable
11. <u>Control Room Toxic Gas (Chlorine)</u>	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	16 (NOTE 5)
12. <u>Control Room Toxic Gas (Ammonia)</u>	
a. TGIS	
(1) Control Room Ventilation - Isolation Mode	36 (NOTE 5)

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps 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 two auxiliary pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Testing the turbine driven pump and both motor driven pumps pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for the turbine driven pump for entry into MODE 3.

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 its correct position.

3. Verifying that both manual valves in the suction lines from the primary AFW supply tank (condensate storage tank T-121) to each AFW pump, and the manual discharge line valve of each AFW pump are locked in the open position.

JUL 16 1982

DESCRIPTION OF PROPOSED CHANGE NDF-10-51 AND SAFETY ANALYSIS
AMENDMENT APPLICATION NO.15 , OPERATING LICENSE NPF-10

This is a request to make various editorial changes in Appendix A, Technical Specifications.

Existing Specifications

See Attachment A.

Proposed Specifications

Page

I	Change spelling for Section 1.12 in Index from "FEATURE" to "FEATURES"
VIII	Delete the last line on this page.
IX	Change the title in Section 3/4.12.3 to "INTERLABORATORY COMPARISON PROGRAM."
X	Change the page number for Section 3/4.3.3 to B 3/4 3-2.
XI	Change the page number for Section 3/4.4.5 to B 3/4 4-4.
XII	Change the page number for Section 3/4.6.4 to B 3/4 6-5.
XIII	Change the title of Section 3/4.7.9 to "FIRE RATED ASSEMBLIES."
XIV	Delete last line.
XVIII	Delete last line.
1-3	Change words in parenthesis of the third line in Section 1.11 to "(in MeV)."
3/4 1-5	Delete footnote from the Applicability Statement of Technical Specification 3.1.1.4 which reads "*See Special Test Exception 3.10-6."
3/4 2-2	Correct the underlining in the title.
3/4 3-3	Delete "@" from Item 17 of Table 3.3-1 of Technical Specification 3.3.1.
3/4 3-4	Delete footnote "@".

3/4 3-8 Delete "###" from Item 10.f of Table 3.3-2 of Technical Specification 3.3.1.

3/4 3-9 Delete "@" from Item 17 of Table 3.3-2.

Delete footnote "@".

Change footnote "##" to read as follows:

> ## Based on a resistance temperature detector (RTD) response time of less than or equal to 6.0 seconds when the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

Delete footnote "###".

3/4 3-10 Add channel calibration footnote "R(4)" to Item 2 of Table 4.3-1 of Technical Specification 4.3.1.1.

3/4 3-11 Delete "@" from Item 17 of Table 4.3-1.

3/4 3-12 Delete footnote "@".

3/4 3-17 Change title to all CAPS for Item 9.

Change title to all CAPS for Item 10.

3/4 3-18 Change title to all CAPS for Item 11.

Change title to all CAPS for Item 12.

3/4 3-24 Change title to all CAPS for Item 9.

Change title to all CAPS for Item 10.

3/4 3-25 Change title to all CAPS for Item 11.

Change title to all CAPS for Item 12.

3/4 3-28 Delete the "a." in front of SIAS in Item 2 of Table 3.3-5.

3/4 3-29 Delete the "a." in Items 5 through 12 of Table 3.3-5.

3/4 3-32 Change title to all CAPS for Item 9.

Change title to all CAPS for Item 10.

3/4 3-33 Change title to all CAPS for Item 11.

Change title to all CAPS for Item 12.

3/4 3-53 Change the ACTION Statements in Item 19 of Table 3.3-10 of Technical Specification 3.3.3.6 from "22, 23" to "20, 21".

Change the ACTION Statement in Item 20 of Table 3.3-10 from "22" to "20".

Change the ACTION Statement in Item 21 of Table 3.3-10 from "22" to "20".

Change the ACTION Statements in Item 22 of Table 3.3-10 from "22, 23" to "22".

3/4 3-53a Change ACTION Statement 22 of Table 3.3-10 to read as follows:

ACTION 22 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

Delete ACTION Statement 23.

3/4 3-56 Delete last set of parenthesis from Part a of the ACTION Statement of Technical Specification 3.3.3.7.

Delete Part b of the ACTION Statement.

Change Part c of the ACTION Statement to Part b.

3/4 3-57 Change Room 301 in Zone 9 of Table 3.3-11 to Room 302.

3/4 3-58 Change Control Building Elevation in Zone 13A of Table 3.3-11 from 30' to 50'.

3/4 3-67 Change footnote in table notation of Table 4.3-8 of Technical Specification 4.3.3.8.1 to read as follows:

*If the instrument controls are not in the operate mode, procedures shall require that the channel be declared inoperable.

3/4 3-70 Delete the words "(See Note 1.)" from ACTION Statement 38 in Table 3.3-13.

3/4 3-70a Delete this page.

3/4 3-74 Delete second line of title which reads 3/4.3.3 "MONITORING INSTRUMENTATION"

3/4 3-75 Delete Technical Specification 4.3.4.1.

Renumber Technical Specification 4.3.4.2 to 4.3.4.

3/4 4-2 Change Technical Specification 3.4.1.2 to read as follows:

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

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

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

3/4 4-3 Change Technical Specification 3.4.1.3 to read as follows:

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

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

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

c. Shutdown Cooling Train A,

d. Shutdown Cooling Train B.

3/4 4-5 Delete "," in Applicability Statement.

3/4 4-6 Delete "S" in word MODES in Applicability Statement.

3/4 4-27 Change the reference in Technical Specification 4.4.8.1.2.b to read as follows:

. . . . Regulatory Guide 1.99, Revision 1, April 1977, ...

3/5 4.29 Change title of Figure 3.4-2 to all CAPS and delete ": SCE 11" from title.

3/4 4-30 Change title of Figure 3.4-3 to all CAPS and delete ": SCE 11" from title,

3/4 5-4 Change RCS pressure in Technical Specification 4.5.2.d.1 from "700 psia" to "715 psia".

3/4 6-6 Change 4.6.1.3.a to read as follows:

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 L_a when determined by flow measurement, with the volume between the door seals pressurized to 9.5 + 0.5 psig for at least 15 minutes,

3/4 6-13 Change Technical Specification 4.6.1.7.1 to read as follows:

4.6.1.7.1 The 42-inch containment purge supply and exhaust isolation valves shall be verified sealed closed at least once per 31 days.

3/4 6-18 Change Technical Specification 4.6.3.2.a to read as follows:

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

3/4 6-21 Add the following valve to Table 3.6-1:

Penetration Number	Valve Number	Function	Max. Isolation Time (Sec.)
30B	HV-7816	Containment air radioactivity monitor outlet	1

3/4 6-22 Change valve number 2"-099-C-376* to 2"-099-C-334*.

Change valve number 2"-321-C-396* to 2"-321-C-376*.

Change penetration number 10B to 10C.

Change valve number 2"-037-C-145 to 2"-037-C-387.

Change valve number 2"-038-C-145 to 2"-038-C-387.

3/4 7-13 Add "*" to title of Technical Specification 3/4 7.5 Control Room Emergency Air Cleanup System.

Change first line in Action Statement to read:

Unit 2 or 3 in MODES 1, 2, 3 or 4:

Change sixth line in Action Statement to read:

Units 2 and 3 in MODES 5 or 6:

Delete the "*" from Part c of the Action Statement.

Change the footnote to read:

*Shared system with San Onofre Unit 3.

3/4 7-26 Change Action Statement Parts a and b of Technical Specification 3.7.8.1 to read as follows:

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

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

Delete Part c of the Action Statement.

3/4 7-29 Change Action Statement Part a of Technical Specification 3.7.8.2 to read as follows:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas outside containment in which redundant systems or components could be damaged; for other areas outside containment, establish an hourly fire watch patrol.

3/4 7-34 Change the last part of the first sentence in Technical Specification 3.7.9 to read as follows:

...., fire dampers, cable, ventilation duct, and piping penetration seals) shall be OPERABLE.

Delete the second sentence in Action Statement part a of Technical Specification 3.7.9.

Add Part e to Technical Specification 4.7.9.1 which reads as follows:

e. Performing a functional test at least once per 18 months of the automatic hold-open, release, closing mechanisms and latches.

3/4 7-35 Change Technical Specification 4.7.9.2 to read as follows:

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices other than fire doors shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/and associated hardware.

c. Performing a visual inspection of at least 10% of each type (mechanical and electrical) of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected at least once per 15 years.

3/4 8-4 Change the units in the first line of Technical Specification 4.8.1.1.2.d.3 from "kw" to "kW".

3/4 10-1 Change Applicability Statement for Technical Specification 3.10.1 to read as follows:

MODES 2 and 3*

Add footnote which reads as follows:

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

3/4 10-2 Change the first sentence in Technical Specification 3.10.2 to read as follows:

3.10.2 The moderator temperature coefficient group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels . . . . .

Change the Action Statement to read as follows:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 and the Minimum Channels . . . . .

Change Technical Specification 4.10.2.1 to read as follows:

The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels . . . . . .

Change Technical Specification 4.10.2.2 to read as follows:

. . . . . the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, 3.2.7 or the Minimum Channels . . . . . .

3/4 10-8 Delete Technical Specifications 3.10.6, 4.10.6.1 and 4.10.6.2.

3/4 11-7 Change Part b of the Action Statement of Technical Specification 3.11.1.4 to read as follows:

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

3/4 11-15 Change Part c of the Action Statement of Technical Specification 3.11.2.5 to read as follows:

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

3/4 12-11 Delete "**" from Applicability Statement of Technical Specification 3.12.2.

3/4 12-12 Delete footnote from Technical Specification 3.12.3.

B3/4 1-3 Change the first paragraph of this page to read:

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) Vortexing, internal structures and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

B3/4 4-1 Change the wording in the first sentence of Section 3/4.4.1 of the Bases to read as follows:

. . . . , and maintain DNBR greater than 1.20 during all normal operations and transients.

B3/4 4-7 Change the last part of the first sentence to read as follows:

. . . . . . for any heatup rate of up to 60°F/hr or cooldown rate of up to 100°F/hr.

Change the tenth line in the third paragraph to read as follows:

the vessel wall by means of the Lead Factor.

B3/4 4-9 Change the tenth line in the first paragraph to read as follows:

HPSI pumps injecting into a water-solid RCS with full charging capacity . . . .

B3/4 5-3 Replace the first paragraph with the following:

The water volume limits are specified relative to the top of the highest suction connection to the tank. (Water volume below this datum is not considered recoverable for purposes of this specification.) The specified volume limits consist of the minimum volume required for ECCS injection above the Recirculation Actuation Signal (RAS) setpoint, plus the minimum volume required for the transition to ECCS recirculation below the RAS setpoint, plus the volume corresponding to the range of the RAS setpoint, including RAS instrument error high and low. Vortexing, internal structure, and instrument error are considered in determining the tank level corresponding to the specified water volume limits.

B3/4 7-2 Replace Section 3/4.7.1.3 with the following:

The OPERABILITY of the condensate storage tank T-121 with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 2 hours followed by cooldown to shutdown cooling initiation, with steam discharge to atmosphere with concurrent loss of offsite power and most limiting single failure. The OPERABILITY of condensate storage tank T-120 in conjunction with tank T-121 ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 24 hours including cooldown to shutdown cooling initiation, with steam discharge to atmosphere with concurrent loss of offsite power and most limiting single failure. The contained water volume limits are specified relative to the highest auxiliary feedwater pump suction inlet in the tank for T-121, and to the T-121 cross connect siphon inlet for T-120. (Water volume below these datum levels is not considered recoverable for purposes of this specification.) Vortexing, internal structure and instrument error are considered in determining the tank levels corresponding to the specified water volume limits.

Prior to achieving 100% RATED THERMAL POWER, Figure 3.7-1 is used to determine the minimum required water volume for T-121 for the maximum power level (hence maximum decay heat) achieved.

B3/4 9-2 Replace Section 3/4.9.6 with the following:

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

With the exception of the four finger CEA's, CEA's are removed from the reactor vessel along with the fuel bundle in which they are inserted utilizing the refueling machine. The four finger CEA's are inserted through the upper guide structure with two fingers in each of two adjacent fuel bundles in the periphery of the core. The four finger CEA's are either removed with the upper guide structure and lift rig or can be removed with separate tooling prior to upper guide structure removal utilizing the auxiliary hoist of the polar crane.

B3/4 10-1 Replace Section 3/4.10-1 with the following:

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

B3/4 10-2 Delete this page.

6-4 Change Table 6.2-1 of Technical Specification 6.2.2 as follows:

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITON

WITH UNIT 3 IN MODE 5 OR 6 OR DE-FUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	2[b]
STA	1	None

WITH UNIT 3 IN MODE 1, 2, 3 or 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
SS	1[a]	1[a]
SRO	1[a]	None
RO	2[b]	1
AO	2[b]	1
STA	1[a]	None

a/ Individual may fill the same position on Unit 3

b/ One of the two required individuals may fill the same position on Unit 3

SS - Shift Supervisor with a Senior Reactor Operators License on Units 2 and 3
SRO - Individual with a Senior Reactor Operators License on Units 2 and 3
RO - Individual with a Reactor Operators License on Units 2 and 3
AO - Auxiliary Operator
STA - Shift Technical Advisor

The last 2 paragraphs of Table 6.2-1 remain unchanged.

6-5 In Technical Specification 6.2.3.2, change the second sentence to read as follows:

> Each shall have a Bachelor's Degree in Engineering or Physical Science or equivalent and at least two years professional level experience in his field.

Add Technical Specification 6.2.3.5 which reads as follows:

> RECORDS
>
> 6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the NSG Supervisor.

6-11 Delete Parts e and f from Technical Specification 6.5.3.5 and reletter Parts g, h, i and j to e, f, g and h respectively.

6-14 Delete the note after Part i of Technical Specification 6.8.1.

6-19 Change the address in Technical Specification 6.9.1.10 from "Office of Management and Program Analysis" to "Office of Resource Management."

Change the title in the third line of Technical Specification 6.9.1.12 from "Director of the Regional Office" to " NRC Regional Administrator."

6-20 Add Parts j, k and l to Technical Specification 6.9.1.12 as follows:

> j. Offsite releases of radioactive materials in liquid and gaseous effluents that exceed the limits of Specifications 3.11.1.1 or 3.11.2.1.
>
> k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.
>
> l. Failure of one or more pressurizer safety valves.

7-1 Delete this page.

7-2 Delete this page.

Reason for Proposed Change

The various editorial changes and corrections in this proposed change are requested to minimize differences between the Unit 2 and proposed Unit 3 Technical Specifications.

Safety Analysis

Corrections contained in Proposed Change NPF-10-51 are editorial or typographical and do not change the intent of the Technical Specifications.

Accordingly, it is concluded that: (1) Proposed Change NPF-10-51 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 or the environment as described in the NRC Final Environmental Statement.

GvN:6119/0213U

ATTACHMENT A

DEFINITIONS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.6 CONTAINMENT SYSTEMS

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ADMINISTRATIVE CONTROLS

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 520°F.

APPLICABILITY: MODES 1 and 2#.*

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 520°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 520°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 535°F.

#With K_{eff} greater than or equal to 1.0.
*See Special Test Exception 3.10.6.

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

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With a F^m_{xy} exceeding a corresponding F^c_{xy}, within 6 hours either:

a. Adjust the CPC and COLSS addressable constants to increase the multiplier applied to PLANAR RADIAL PEAKING FACTORS to a factor greater than or equal to (F^m_{xy}/F^c_{xy}); or

b. Adjust only the CPC addressable constants as in (a). Restrict subsequent operation so that a margin tc the COLSS operating limits of at least [(F^m_{xy}/F^c_{xy}) - 1.0] x 100% is maintained; or

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

d. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

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

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

b. At least once per 31 EFPD.

*See Special Test Exception 3.10.2.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2 sets of 2 2 sets of 2	1 set of 2 1 set of 2	2 sets of 2 2 sets of 2	1, 2 3*, 4*, 5*	1 7A
2. Linear Power Level - High	4	2	3	1, 2	2#, 3#
3. Logarithmic Power Level-High					
a. Startup and Operating	4 4	2(a)(d) 2	3 3	1, 2 3*, 4*, 5*	2#, 3# 7A
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	4	2	3	1, 2 3*, 4*, 5*	2#, 3# 7A
13. Reactor Trip Breakers	4	2(f)	4	1, 2 3*, 4*, 5*	5 7A
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3#, 7
15. CEA Calculators	2	1	2(e)	1, 2	6, 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
@17. Seismic - High	4	2	3	1, 2	2#, 3#
18. Loss of Load	4	2	3	1(g)	2#, 3#

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 5% 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 6.5.1.6e. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	$\leq$ 0.40 seconds*
3. Logarithmic Power Level - High	$\leq$ 0.45 seconds*
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Pressurizer Pressure - Low	$\leq$ 0.90 seconds
6. Containment Pressure - High	$\leq$ 0.90 seconds
7. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
8. Steam Generator Level - Low	$\leq$ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 0.68 seconds*
b. CEA Positions	$\leq$ 0.68 seconds**
c. CEA Positions: CEAC Penalty Factor	$\leq$ 0.53 seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 0.68 seconds*
b. CEA Positions	$\leq$ 0.68 seconds**
c. Cold Leg Temperature	$\leq$ 0.68 seconds##
d. Hot Leg Temperature	$\leq$ 0.68 seconds##
e. Primary Coolant Pump Shaft Speed	$\leq$ 0.68 seconds#
f. Reactor Coolant Pressure from Pressurizer	$\leq$ 0.68 seconds###
g. CEA positions: CEAC Penalty Factor	$\leq$ 0.53 seconds

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow-Low	0.9 sec
@17. Seismic-High	Not Applicable
18. Loss of Load	Not Applicable

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*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Response time shall be measured from the output of the sensor. RTD response time shall be measured at least once per 18 months by means of the Loop Current Step Response (LCSR) method. The measured R_T of the slowest RTD shall be less than or equal to 6.0 seconds.

###Response time shall be measured from the output of the pressure transmitter. The transmitter response time constant shall be less than or equal to 0.7 seconds where the pressure transmitter response time is equivalent to the time interval required for the transmitter to achieve 63.2% of its total change when subjected to a step change in pressure transmitter pressure.

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

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)

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.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
9. Control Room Isolation (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13*#
b. Manual SIAS (Trip Buttons)	2 sets of 2/unit	1 set of 2	2 sets of 2/unit	1, 2, 3, 4	8
c. Airborne Radiation					
i. Particulate/Iodine	2	1	1	All	13*#
ii. Gaseous	2	1	1	All	13*#
d. Automatic Actuation Logic	1/train	1	1	All	13*#
10. Toxic Gas Isolation (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*#, 15*#
b. Chlorine - High	2	1	1	All	14*#, 15*#
c. Ammonia - High	2	1	1	All	14*#, 15*#
d. Butane/Propane - High	2	1	1	All	14*#, 15*#
e. Carbon Dioxide - High	2	1	1	All	14*#, 15*#
f. Automatic Actuation Logic	1/train	1	1	All	14*#, 15*#

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
11. Fuel Handling Isolation (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16*#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16*#
ii. Particulate/Iodine	2	1	1	**	16*#
c. Automatic Actuation Logic	1/train	1	1	**	16*#
12. Containment Purge Isolation (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17*#
b. Airborne Radiation					
i. Gaseous	2	1	1	All	17, 17a, 17b
ii. Particulate	2	1	1	All	17, 17a, 17b
iii. Iodine	2	1	1	All	17, 17b
c. Containment Area Radiation (Gamma)	2	1	1	6	17*#
d. Automatic Actuation Logic	1/train	1	1	All	17, 17a, 17b*#

TABLE 3.[illegible] (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. Control Room Isolation (CRIS)		
a. Manual CRIS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Airborne Radiation		
i. Particulate/Iodine	$\leq 5.7 \times 10^4$ cpm**	$\leq 6.0 \times 10^4$ cpm**
ii. Gaseous	$\leq 3.8 \times 10^2$ cpm**	$\leq 4.0 \times 10^2$ cpm**
d. Automatic Actuation Logic	Not Applicable	Not Applicable
10. Toxic Gas Isolation (TGIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Chlorine - High	$\leq$ 6.0 ppm	$\leq$ 6.2 ppm
c. Ammonia - High	$\leq$ 42.4 ppm	$\leq$ 44.7 ppm
d. Butane/Propane - High	$\leq$ 84.8 ppm	$\leq$ 89.3 ppm
e. Carbon Dioxide - High	$\leq$ 4061.3 ppm	$\leq$ 4275.0 ppm
f. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. Fuel Handling Isolation (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq 1.3 \times 10^{2}$ cpm**	$\leq 1.4 \times 10^{2}$ cpm**
ii. Particulate/Iodine	$\leq 5.7 \times 10^{4}$ cpm**	$\leq 6.0 \times 10^{4}$ cpm**
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. Containment Purge Isolation (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	$\leq$ per ODCM	$\leq$ per ODCM
ii. Particulate	$\leq$ per ODCM	$\leq$ per ODCM
iii. Iodine	$\leq$ per ODCM	$\leq$ per ODCM
c. Containment Area Radiation (Gamma)	$\leq$ 2.4 mR/hr	$\leq$ 2.5 mR/hr
d. Automatic Actuation Logic	Not Applicable	Not Applicable

Table 3.3-5 (continued)

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME (SEC)
2. Pressurizer Pressure-Low	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	31.2*
(b) Low Pressure Safety Injection	41.2*
(2) Control Room Isolation	Not Applicable
(3) Containment Isolation (NOTE 3)	11.2* (NOTE 2)
(4) Containment Spray (Pumps)	25.6*
(5) Containment Emergency Cooling	
(a) CCW Pumps	31.2*
(b) CCW Valves (Note 4a)	21.2
(c) CCW Valves (Note 4b)	23.2*
(d) Emergency Cooling Fans	21.2*
3. Containment Pressure-High	
a. SIAS	
(1) Safety Injection	
(a) High Pressure Safety Injection	41.0*
(b) Low Pressure Safety Injection	41.0*
(2) Control Room Isolation	Not Applicable
(3) Containment Spray (Pumps)	25.4*
(4) Containment Emergency Cooling	
(a) CCW Pumps	31.0*
(b) CCW Valves (Note 4a)	21.0
(c) CCW Valves (Note 4b)	23.0*
(d) Emergency Cooling Fans	21.0*
b. CIAS	
Containment Isolation	10.9* (NOTE 2)
4. Containment Pressure - High-High	
CSAS	
Containment Spray	21.0*

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)	50.9*/40.9**
(2) Auxiliary Feedwater (steam/DC train)	30.9 (Note 6)
9. Steam Generator Level - Low (and ΔP - High)	
a. EFAS	
(1) Auxiliary Feedwater (AC trains)	50.9*/40.9**
(2) Auxiliary Feedwater (Steam/DC train)	30.9 (Note 6)
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 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER (LOV)				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	S	R	R	1, 2, 3, 4
8. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)(3), SA(4)	1, 2, 3
9. Control Room Isolation (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	R(3)	All
10. Toxic Gas Isolation (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Carbon Dioxide - High	S	R	M	All
f. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Fuel Handling Isolation (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R(3)	*
12. Containment Purge Isolation (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	(2)	(2)	(2)	All
ii. Particulate	(2)	(2)	(2)	All
iii. Iodine	(2)	(2)	(2)	All
c. Containment Area Radiation (Gamma)	S	R	M	6
d. Automatic Actuation Logic	N.A.	N.A.	R (3)	All

TABLE NOTATION

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

(2) In accordance with Table 4.3-9 surveillance requirements for these instrument channels.

(3) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(4) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay.

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

* With irradiated fuel in the storage pool.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (CONTINUED)

	INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
17.	Containment Water Level - Wide Range	2	1	20, 21
18.	Core Exit Thermocouples	7/core quadrant	4/core quadrant	20, 21
19.	Containment Area Radiation - High Range	2	1	22, 23
20.	Main Steam Line Area Radiation	1/steam line	N.A.	22
21.	Condenser Evacuation System Radiation Monitor - Wide Range	1	N.A.	22
22.	Purge/Vent Stack Radiation Monitor - Wide Range*	2	1	22, 23
23.	Cold Leg HPSI Flow	1/cold leg	N.A.	20
24.	Hot Leg HPSI Flow	1/hot leg	N.A.	20

NOTES:

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

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 20 - With the number of OPERABLE accident monitoring channels less than the Required Number of Channels, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

ACTION 21 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 22 - With the number of channels OPERABLE less than the Required Number of Channels, comply with the ACTION requirements of Specification 3.3.3.6.

ACTION 23 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

1) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.7 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

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

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5).

b. Restore the inoperable instrument(s) to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.7.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.7.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.7.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

4.3.3.7.4 Following a seismic event (basemat acceleration greater than or equal to 0.05 g):

a. Within 2 hours each zone shown in Table 3.3-11 shall be inspected for fires, and

b. Within 72 hours an engineering evaluation shall be performed to verify the OPERABILITY of the fire detection system in each zone shown in Table 3.3-11.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"			10			
	Cable Tray Areas Elev 45'			9			
	Cable Tray Areas Elev 30'			4			
	Elevator Machinery Room			1			
	Combustible Oil Area Two steam generator rooms				32		
	Charcoal Filter Area Elev 45'	2					
2	Penetration						
	Elev 63'6"			12			
4	New Fuel Storage Area and Spent Fuel Pool Areas						
	Spent Fuel Pool		4				
	New Fuel Pool		3				
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2	24		
	Cable Riser Gallery Rm 449			3	24		
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank Rms 511A		None				
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A	1		1			
	Emgy. A.C. Unit Rm 301-Train B	1		1			
10	Penetration						
	Elev 45'			6			

*The fire detection instruments located within the Containment are not required to be OPERABLE during the performance of Type A Containment Leakage Rate Tests.

TABLE 3.3-11 (Continued)

Zone	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
11	S.E.B. Roof and Main Steam Relief Valves	None					
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3	42		
	Cable Riser Gallery Rm 315			3	40		
13A	Control Building Elev 30'						
	Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 50'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50'						
	Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3		4	
	Train B			3		4	
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2		6	
23	Fuel Handling Bldg Elev 30'						
	Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'	2					

TABLE 4.3-8 (Continued)

TABLE NOTATION

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:*

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

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

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

*If the instrument controls are not in the operate mode, procedures shall call for declaring the channel inoperable.

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas holdup system operation (treatment for primary system offgases).

a) In accordance with Table 3.3-6 ACTION 19

b) In accordance with the ACTION Requirements of Specification 3.4.5.1 (Modes 1, 2, 3 and 4)

c) In accordance with the ACTION Requirement of Specification 3.9.9 (Mode 6)

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway. (See Note 1.)

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With two channels inoperable, be in at least HOT STANDBY within 6 hours.

ACTION 40 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 3.3-13 (Continued)

TABLE NOTATION

Note 1

From August 6, 1982, through September 6, 1982, containment purge with Noble Gas Activity Monitor 2RT-7804-1 inoperable is permissable for no more than two hours per day provided that:

(1) Vent stack monitor 2RT-7865-1 is OPERABLE and aligned to the purge stack for the duration of the purge.

(2) In the event of a high activity alarm on 2RT-7865-1 during the purge, an operator will (a) suspend containment purge and then (b) realign 2RT-7865-1 to the vent stack.

(3) When purging is completed, 2RT-7865-1 is returned to its normal alignment.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more loose part detection system channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the loose-part detection system shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 24 hours,

b. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

c. CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one turbine overspeed protection system shall be OPERABLE.

APPLICABILITY: MODES 1, 2* and 3.*

ACTION:

a. With one stop valve or one control valve per high pressure turbine steam lead inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam lead inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lead or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required turbine overspeed protection system shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position.

1. Four high pressure turbine stop valves.
2. Four high pressure turbine control valves.
3. Six low pressure turbine reheat stop valves.
4. Six low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position.

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the turbine overspeed protection systems.

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.

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*With any main steam line isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The Reactor Coolant loops 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.

b. At least one of the above Reactor Coolant loops shall be in operation.*

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

*All Reactor Coolant 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.

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.

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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

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling train shall be OPERABLE 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.

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REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 with Reactor Coolant loops not filled.

ACTION:

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

b. With no shutdown cooling trains 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

4.4.1.4.2 At least one 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.

REACTOR COOLANT SYSTEM

3/4.4.8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.8.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4-2 and Figure 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 30°F in any one hour period with RC cold leg temperature less than 280°F. A maximum heatup of 60°F in any one hour period with RC cold leg temperature greater than 280°F.

b. A maximum cooldown of 30°F in any one hour period with RC cold leg temperatures less than 280°F. A maximum cooldown of 100°F in any one hour period with RC temperature greater than 280°F.

c. A maximum temperature change of less than or equal to 10°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

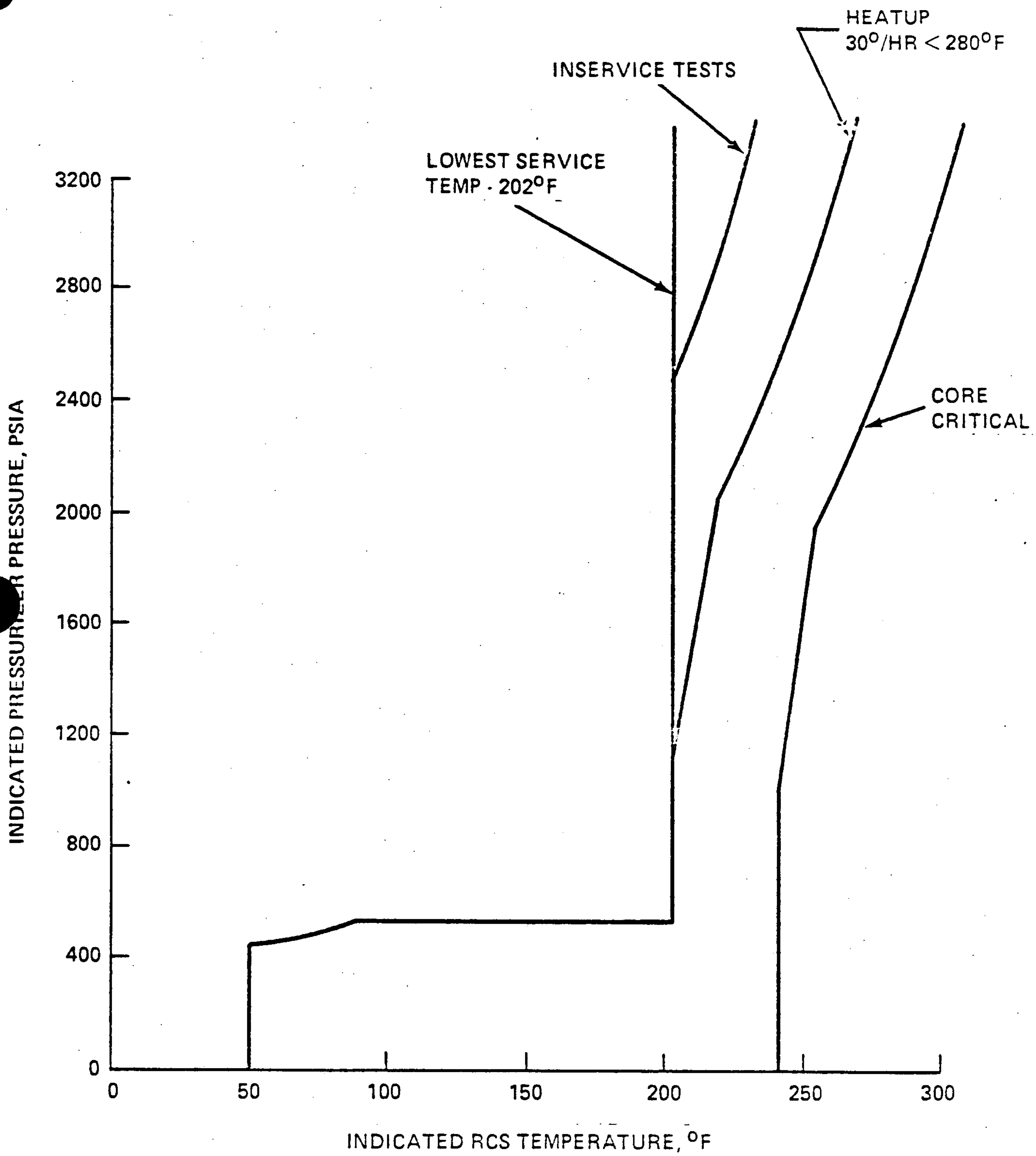
4.4.8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3. Recalculate the Adjusted Reference Temperature based on the greater of the following:

a. The actual shift in reference temperature for plates C-6404-2 as determined by impact testing, or

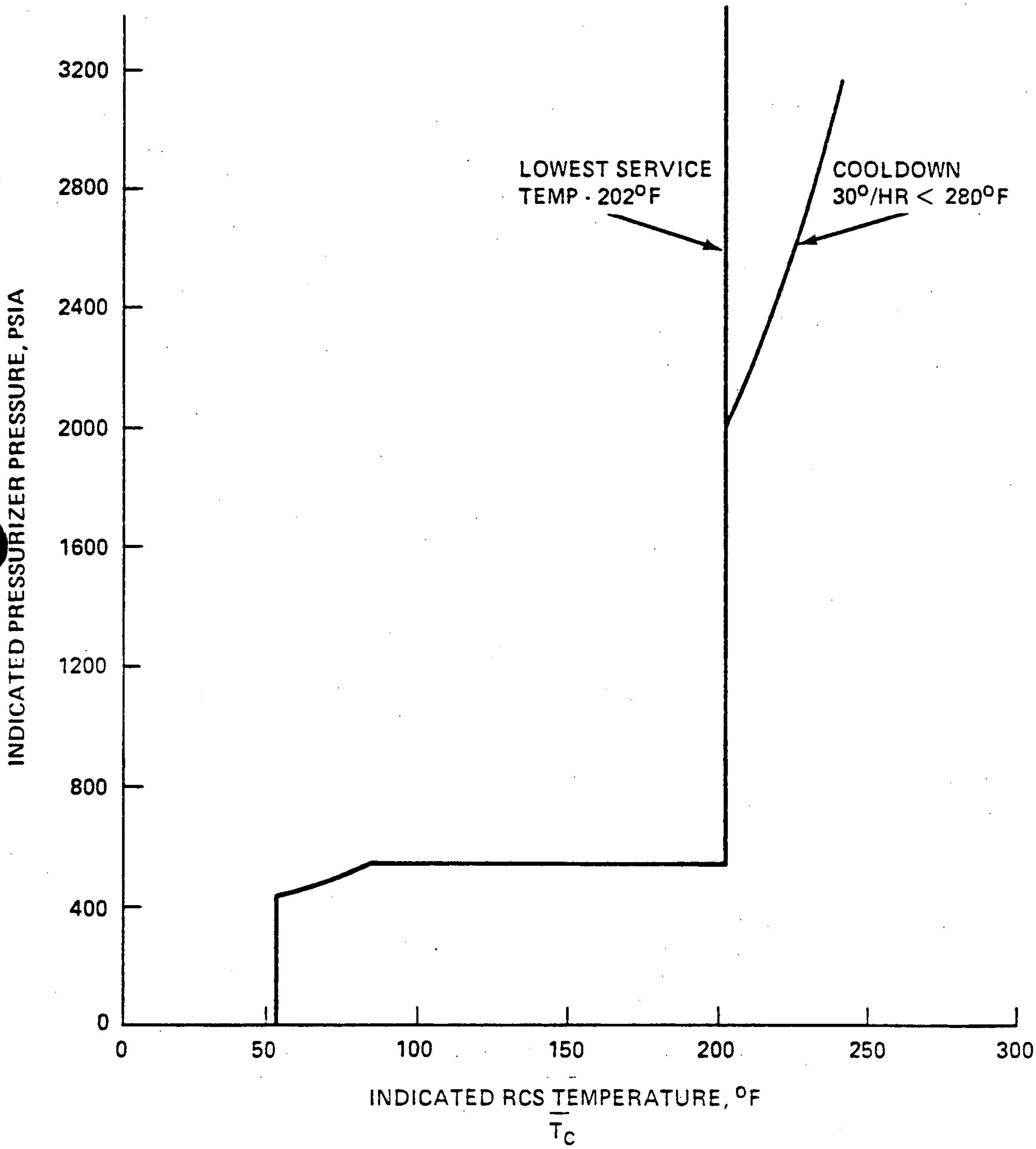
b. The predicted shift in reference temperature for weld seams 3-203A or 3-203B as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Figure 3.4-2



Heatup RCS pressure/temperature limitations for: SCE II 0-5 years.

Figure 3.4-3



Cooldown RCS pressure/temperature limitations for: SCE II 0-5 years.

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.

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CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Containment purge supply and exhaust isolation valves shall be OPERABLE and:

a. Each 42-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. Each 8-inch containment purge supply and exhaust isolation valve may be open for less than or equal to 1000 hours per 365 days.

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

ACTION:

a. With the 42-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, or with the 8-inch purge supply and/or exhaust isolation valve(s) open for more than 1000 hours per 365 days, close and/or seal closed the open valve(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a 42-inch or 8-inch containment purge supply and/or exhaust isolation valve having a measured leakage rate exceeding the limits of Surveillance Requirement 4.6.1.7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SUREVILLANCE REQUIREMENTS

4.6.1.7.1 The 42-inch containment purge supply and exhaust isolation valves shall be verified to be:

a. Closed at least once per 24 hours.

b. Sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative time that the 8-inch purge supply and exhaust isolation valves are open during the past 365 days shall be determined at least once per 7 days.

4.6.1.7.3 At least once per 3 months each 42 inch and each 8 inch purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.6.3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

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

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

TABLE 3.6-1 (Continued)

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
28	HV-4052#	Steam generator feedwater	10
29	HV-4048#	Steam generator feedwater	10
30A	HV-7802	Containment air radioactivity monitor inlet	1
30A	HV-7803	Containment air radioactivity monitor inlet	1
30B	HV-7801	Containment air radioactivity monitor outlet	1
30B	HV-7800	Containment air radioactivity monitor outlet	1
30C	HV-0516	Quench tank and drain tank gas sample	40
30C	HV-0514	Quench tank and drain tank gas sample	40
30C	HV-0515	Quench tank and drain tank gas sample	40
32	HV-8204#	Mainsteam isolation	5
33	HV-8205#	Mainsteam isolation	5
42	HV-6211	Component cooling water inlet	40
43	HV-6216	Component cooling water outlet	40
45	HV-9900	Containment normal A/C chilled water inlet	40
45	HV-9920	Containment normal A/C chilled water inlet	40
46	HV-9971	Containment normal A/C chilled water inlet	40
46	HV-9921	Containment normal A/C chilled water outlet	40
47	HV-7258	Containment waste gas vent header	40
47	HV-7259	Containment waste gas vent header	40
77	HV-5434	Nitrogen supply to safety injection tanks	40
B. CONTAINMENT PURGE (CPIS)			
18	HV-9949**	Containment purge inlet (normal)	12
18	HV-9948**	Containment purge inlet (normal)	12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950**	Containment purge outlet (normal)	12
19	HV-9951**	Containment purge outlet (normal)	12
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

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

3/4.7.5 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 Two independent control room emergency air cleanup systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

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

MODES 5 and 6:

a. With one control room emergency air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency air cleanup system in the recirculation mode.

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

c. The provisions of Specification 3.0.3 are not applicable in MODE 6.*

SURVEILLANCE REQUIREMENTS

4.7.5 Each control room emergency air cleanup system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 110°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on.

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

   1. Verifying that with the system operating at a flow rate of 35485 cfm ± 10% for the air conditioning unit, and 1000 cfm ± 10% for the ventilation unit and recirculating through the respective HEPA filters and charcoal adsorbers, leakage through the system diverting valves is less than or equal to 1% air conditioning unit and 1% ventilation unit when the system is tested by admitting cold DOP at the respective intake.

*Specification 3.0.4 not applicable for initial entry into MODE 6.

3/4.7.8 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 The fire suppression water system shall be OPERABLE with:

a. Two electric motor-driven fire pumps, each with a capacity of 1500 gpm and one diesel-driven fire pump with a capacity of 2500 gpm, with their discharge aligned to the fire suppression header,

b. Two separate water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path capable of taking suction from each water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the first valve upstream of the water flow alarm device on each spray and/or sprinkler or fire hose station required to be OPERABLE per Specifications 3.7.8.2 and 3.7.8.3.

APPLICABILITY: At all times.

ACTION:

a. With one required electric motor-driven/diesel-driven pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

   1. Establish a backup fire suppression water system within 24 hours, and

   2. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2:

      a) By telephone within 24 hours,

      b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas outside containment in which redundant systems or components could be damaged; for other areas outside containment, establish an hourly fire watch patrol. Restore the system to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

b. With one or more of the above required spray and/or sprinkler systems inside containment inoperable, restore the system to OPERABLE status within 24 hours or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 7 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) outside of containment in the flow path is in its correct position.

b. At least once per 31 days during each COLD SHUTDOWN or REFUELING by verifying that each valve (manual, power operated or automatic) inside containment in the flow path is in its correct position,

c. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures and other fire barriers) separating safety related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable and piping penetration seals and ventilation seals) shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within one hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol. Restore the inoperable fire rated assembly and sealing device to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperable fire rated assembly and/or sealing device and the plans and schedule for restoring the fire rated assembly and sealing device to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1 Each of the above required fire doors shall be verified OPERABLE by:

a. Verifying at least once per 24 hours the position of each closed fire door and that doors with automatic hold-open and release mechanisms are free of obstructions.

b. Verifying at least once per 7 days the position of each locked closed fire door.

c. Performing a CHANNEL FUNCTIONAL TEST at least once per 31 days of the fire door supervision system.

d. Inspecting at least once per 6 months the automatic hold-open, release and closing mechanism and latches.

SURVEILLANCE REQUIREMENTS

4.7.9.2 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire window/fire damper/ and associated hardware.

c. Performing a visual inspection of at least 10 percent each type (mechanical and electrical) of sealed penetrations. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10 percent of that particular type of sealed penetration shall be made. This inspection process shall continue until an additional complete 10 percent sample of that type of sealed penetration with no apparent changes in appearance or abnormal degradation are found.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the generator capability to reject a load of 4700 kw without tripping. The generator voltage shall not exceed 5450 volts during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the permanently connected loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4360 ± 436 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ESF test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady state generator voltage and frequency shall be 4360 ± 436 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and freqency shall be maintained within these limits during this test.

6. Verifying that on a simulated loss of the diesel generator (with offsite power not available), the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of offsite power in conjunction with an ESF test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto connected emergency (accident) loads through the load sequence and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After loading, the steady state voltage and frequency of the emergency busses shall be maintained at 4360 ± 436 volts and 60 + 1.2/-0.3 Hz during this test.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 2.

ACTION:

a. With any full length CEA not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

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

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

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 may be suspended during the performance of PHYSICS TESTS provided:

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

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

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended, either:

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

b. Be in HOT STANDBY wthin 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

SPECIAL TEST EXCEPTIONS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.6 The minimum temperature for criticality limits of 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 $\leq$ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship and the minimum temperature for criticality are 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 At least once per hour:

a. 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.

b. The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER.

c. The Reactor Coolant System temperature shall be verified to be greater than or equal to 320°F.

4.10.6.2 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.

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LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any outside temporary tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

b. The provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each outside temporary tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within one hour and less than or equal to 2% by volume within 48 hours.

c. The provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.13b are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.

APPLICABILITY: At all times.**

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment via the same exposure pathway 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment via the same exposure pathway may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of June 1 and October 1 using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.

*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

FEB

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.*

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with the ODCM shall be included in the Annual Radiological Environmental Operating Report.

*Interlaboratory comparison program not required prior to first exceeding 5% RATED THERMAL POWER or July 1, 1982, whichever occurs first.

BORATION SYSTEMS (Continued)

The water volume limits are specified relative to the limiting physical characteristics of the tanks and includes allowances for water not available because of discharge line location and other physical characteristics (RWST above the ECCS suction connection in lieu of the CVCS suction connection).

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on water volume and boron concentration of the RWST also ensure a pH value of between 8.0 and 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable, CEA to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

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.

PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup and cooldown limit curves (Figures 3.4-2 and 3.4-3) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 60°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2 and 3.4-3.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper and phosphorous content of the material in question, can be predicted using FSAR Table 5.2-5 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel taking into account the location of the sample closer to the core than the vessel wall by means of the Lead Factor. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 90°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 and 3.4-3 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

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

PRESSURE/TEMPERATURE LIMITS (Continued)

The OPERABILITY of the Shutdown Cooling System relief valve or a RCS vent opening of greater than 5.6 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 235°F. The Shutdown Cooling System relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures or (2) inadvertant safety injection actuation with two HPSI pump and its injection into a water solid RCS with full charging capacity and letdown isolated.

3/4.4.9 STRUCTURAL INTEGRITY

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

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

REFUELING WATER STORAGE TANK (Continued)

The water volume limits are specified relative to the limiting physical characteristics of the tanks and includes allowances for water not available because of discharge line location and other physical characteristics (RWST above the ECCS suction connection in lieu of the CVCS suction connection).

The limits on water volume and boron concentration of the RWST also ensure that the solution recirculated within containment after a LOCA has a pH value between 8.0 and 10.0 at the end of the NaOH injection period. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The maximum RWST volume is not specified since analysis of pH limits and containment flooding post-LOCA considered RWST overflow conditions.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

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

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1170 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 1170 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the condensate storage tanks with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 24 hours with steam discharge to atmosphere with concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

BASES

3/4.9.6 REFUELING MACHINE

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

3/4.9.7 FUEL HANDLING MACHINE - SPENT FUEL STORAGE BUILDING

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

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling train be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

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

3/4.9.9 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while below THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 RADIATION MONITORING/SAMPLING

This special test exception permits fuel loading and reactor operation with radiation monitoring/sampling instrumentation calibration and quality assurance conforming to either FSAR procedures or Regulatory Guide 4.15 Rev 1, February 1979. This test exception is required to allow for a phased implementation of Regulatory Guide 4.15 Rev. 1, February 1979. Equivalent instrumentation, quality assurance and/or calibration is provided until full implementation of Regulatory Guide 4.15 Rev. 1, February 1979.

The containment airborne monitors and associated sampling media test exception is required to allow for operation prior to and during installation of upgraded monitors/media. Adequate monitoring is provided until and subsequent to the completion of the upgraded installation. Extensive containment air mixing during high volume purge (MODES 5 and 6) occurs as a result of containment HVAC and fans resulting in representative air monitoring via either 2RT-7804-1 or 2RT-7807-2. During low volume purge operations (MODES 1, 2, 3 and 4) 2RT-7804-1 provides representative indication of purged air due to its location in the immediate vicinity of the low volume purge exhaust.

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BASES

3/4.10.6 MINIMUM TEMPERATURE FOR CRITICALITY

This special test exception permits reactor criticality at low THERMAL POWER levels with T_{avg} below 520°F during PHYSICS TESTS which provide data that can be used to verify the adequacy of design codes for new fuel designs for reduced temperature conditions. The Low Power Physics Testing program at low temperature (320°F) is used to perform the following tests:

1. Biological shielding survey test
2. Isothermal temperature coefficient tests
3. Regulatory CEA group tests
4. Boron worth tests
5. Critical configuration boron concentration

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION

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

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

During any absence of the Shift Supervisor from the Control Room 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 Shift Supervisor 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.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

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

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five dedicated full-time engineers. Each shall have a Bachelor's Degree in Engineering or Physical Science and at least two years professional level experience in his field. Off-duty qualified Shift Technical Advisors may be used to fulfill this requirement.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving plant safety to the Supervisor, Nuclear Safety Group.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a Bachelor's Degree or equivalent in a scientific or engineering discipline with specific training in plant design and in the response and analysis of the plant for transients and accidents.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

*Not responsible for sign-off function.

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

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

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

g. Events requiring 24 hour written notification to the Commission.

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

i. Reports and meetings minutes of the Onsite Review Committee.

AUDITS

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

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

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

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

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

e. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

g. Any other area of unit operation considered appropriate by the Nuclear Safety Group or Manager of Nuclear Operations.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

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 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.

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 Deputy Station Manager, (2) the Manager, Operations, (3) the Manager, Maintenance, (4) the Manager, Technical, or (5) the Manager, Health Physics 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. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the 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

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The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

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

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the Onsite Review Committee.

REPORTABLE OCCURRENCES

6.9.1.11 The REPORTABLE OCCURRENCES of Specifications 6.9.1.12 and 6.9.1.13 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.12 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup 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. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

7.0 SPECIAL TEST PROGRAM

7.1 For conducting the special low power test program as described in Section 22.2-1.G.1 of Supplement No. 1 to the Safety Evaluation Report (SER) the Technical Specifications may be exempt (E) or modified (C) as follows:

Technical Specifications			Test	Test	Test
Section	Description		A1	A2	A3
2.2.1	Reactor Trip Setpoints				
	2.	Linear Power Level-High Four Reactor Coolant Pumps Operating	C(1)	C(1)	C(1)
	3.	Logarithmic Power Level-High	C(2)	C(2)	C(2)
	5.	Pressurizer Pressure-Low		C(3)	
	7.	Steam Gen. Pressure-Low	C(4)	C(4)	C(4)
	9.	Local Power Density-High	E(5)	E(5)	E(5)
	10.	DNBR-Low	E(5)	E(5)	E(5)
	11.	Reactor Coolant Flow-Low	E(5)	E(5)	E(5)
3.3.1	Reactor Protective Instrumentation				
	9.	Local Power Density-High	E(5)	E(5)	E(5)
	10.	DNBR-Low	E(5)	E(5)	E(5)
	14.	Core Protection Calculators	E(5)	E(5)	E(5)
	16.	Reactor Coolant Flow-Low	E(5)	E(5)	E(5)
3.3.2	Engineered Safety Feature Actuation System Instrumentation				
	1.	Safety Injection (SIAS)		C(3)	
	4.	Main Steam Line Isolation	C(4)	C(4)	C(4)
	6.	Containment Cooling (CCAS)		C(3)	
	8.	Emergency Feedwater (EFAS)	C(4)	C(4)	C(4)

Notes:

1. Trip setpoint lowered to ≤ 9.1% RATED THERMAL POWER, allowable value ≤ 10.4% RATED THERMAL POWER

2. Trip setpoint raised to ≤ 100% RATED THERMAL POWER, allowable value ≤ 100% RATED THERMAL POWER

3. Trip setpoint lowered to ≥ 1,550 psia

4. Trip setpoint lowered to ≥ 550 psia

5. Trip bypassed

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