ATTACHMENT "A"

Evaluation of necessary changes to Unit 1 Technical Specifications as identified by page number.

Page 1-1	The adjective "principal" should be used in the "ACTION" definition.
Page 1-2	The typo "mechanism" should be corrected.
Page 2-9	The adjective "operating" should be used in the footnote.
Page 3/4 1-9* 1-11* 1-12*	For overpressure conditions on ECCS pumps specifications should be changed to agree with Unit 2.
Page 3/4 1-18* 1-19* 1-20* 1-21* 1-22* 1-28*	Part length rods, specifications should be changed to agree with Unit 2.
Page 3/4 3-20* 3-22* 3-26a* 3-29* 3-30* 3-33*	For degraded grid voltage protection, specifications should be changed to agree with Unit 2 requirements.
Page 3/4 3-26	Item 4.d should identify the T_{avg} term as Low-Low rather than Low.
Page 3/4 3-31	Item 1.f should identify the T_{avg} term as Low-Low rather than Low.
Page 3/4 3-33	Item 4.d should identify the T_{avg} term as Low-Low rather than Low.
Page 3/4 3-52	Specification 4.3.3.7.2 should delete reference to "Class A" since circuits do not fit the definition.
Page 3/4 4-26* 4-27* 4-28*	Update figures for heatup and cooldown curves to agree with figures previously submitted as part of the overpressure protection review. Similar to Unit 2, limits for hydrostatic testing are included.
Page 3/4 4-29*	Reactor vessel material irradiation surveillance schedule needs revision to be consistent with Unit 2 specifications and WCAP-8771.
Page 3/4 4-30a* 4-30b* 5-3* 5-6*	Add a new overpressure protection specification and basis for unit which is identical to Unit 2 except for setpoints.
Page 3/4 6-33	The typo "Volume" should be corrected.
Page 3/4 6-35 6-36	The entire surveillance requirement should be changed to agree with the proposed Unit 2 requirement, since the hardware is the same equipment for both Units 1 and 2.

533 062

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Page 3/4 7-20	The typo ", and " should be corrected in specification 3.7.6.1.b.1.
Page 3/4 7-21	The specification for the control room habitability systems should be made consistent with proposed Unit 2 requirements, since the control room is a common area.
Page 3/4 7-24	Make surveillance requirements for safeguards area ventilation systems consistent with Unit 2.
Page 3/4 7-68 7-69	The entire Specification should be changed to agree with the proposed Unit 2 requirements, since the hardware is the same equipment for both Units 1 and 2.
Page 3/4 7-79	The entire Specification should be changed to agree with the proposed Unit 2 requirements, since the hardware is the same equipment for both Units 1 and 2.
Page 3/4 9-10	Remove typographical error.
Page 3/4 10-1*	Part length rod specifications should agree with Unit 2.
Page 3/4 10-2* 10-3*	Remove references to deleted specification.
Page B3/4 1-4*	Part length rod Basis should be changed to agree with Unit 2.
Page B3/4 4-10*	Reactor vessel toughness table has typographical errors which need correction.
Page B3/4 4-11*	Add new Basis for overpressure protection to agree with Unit 2.
Page B3/4 7-7	The entire Basis should be changed to agree with the proposed Unit 2 requirements since the hardware is the same equipment for both Units 1 and 2.
Page 5-4*	Part length rods as a design feature should agree with Unit 2.
Page 6-1	Specification 6.2.2.f should be changed to agree with the proposed Unit 2 specification since it is a requirement for the same people.
Page 6-10	A typo in Specification 6.5.2.7.d.3 should be corrected. Technical Specifications should be capitalized.
Page 6-13	Specification 6.8.1.a should be changed to agree with the proposed Unit 2 specification since Unit 1 and Unit 2 procedures should be consistent.
Page 6-22	Specification 6.12 should be changed to agree with proposed Unit 2 specification since it is a requirement for common areas.

ATTACHMENT B

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2775 MWt.

OPERATIONAL MODE - MODE

1.4 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each ~~principle~~ specification and shall be part of the specifications. principal

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

533 065

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specification 6.9.1.8 and 6.9.1.9.

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

1.8.1 All penetrations required to be closed during accident conditions are either:

a. Capable of being closed by an OPERABLE containment automatic isolation valve system, or

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1,

1.8.2 All equipment hatches are closed and sealed,

1.8.3 Each air lock is OPERABLE pursuant to Specification 3.6.1.3,

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2, and

1.8.5 The sealing mechanism ~~mechansim~~ associated with each penetration (e.g. welds, bellows or O-rings) is OPERABLE.

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

533 066

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Operation with 3 Loops	Operation with 2 Loops (no loops isolated)*	Operation with 2 Loops (1 loop isolated)*
K_1 = 1.141	K_1 = ( )	K_1 = ( )
K_2 = 0.0128	K_2 = ( )	K_2 = ( )
K_3 = 0.000608	K_3 = ( )	K_3 = ( )

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between - 34 percent and + 10 percent, f_1 (ΔI) = 0 (where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER).

(ii) for each percent that the magnitude of ($q_t - q_b$) exceeds - 34 percent, the ΔT trip setpoint shall be automatically reduced by 3 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of ($q_t - q_b$) exceeds + 10 percent, the ΔT trip setpoint shall be automatically reduced by 1.25 percent of its value at RATED THERMAL POWER.

operating

*Values dependent on NRC approval of ECCS evaluation for these operation conditions.

533 067

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

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

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is $\geq$ 145°F.

\# Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

533 068

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one char;ing pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 At least the above required charging pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 2410 psig when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours by verifying that the switches in the Control Room have been placed in the pull to lock position.

533 069

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 No more than two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore a second charging pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1.77% $\Delta k/k$ at 200°F within the next 6 hours; restore a second charging pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours. The provisions of Specification 3.0.4 are not applicable for one hour ~~prior to bubble formation or collapse in MODE 4~~. following heatup above 320°F or prior to cooldown below 320°F.

SURVEILLANCE REQUIREMENTS

4.1.2.4 The above required charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of $\geq$ 2410 psig when tested pursuant to Specification 4.0.5. | 2

~~*With the reactor coolant system solid, no more than one charging pump shall be OPERABLE.~~

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

533 070

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within $\pm$ 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

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

b. With more than one full length rod inoperable or misaligned from the bank step counter demand position by more than $\pm$ 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable due to causes other than those addressed by ACTION "a" above or misaligned from its bank step counter demand height by more than $\pm$ 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days. This reevaluation shall confirm that the previous analyzed results of these accidents remain valid for the duration of operation under these conditions, and

*See Special Test Exceptions 3.10.2 and 3.10.3.

533 071

LIMITING CONDITION FOR OPERATION (Continued)

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

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, or

d) Either the THERMAL POWER level is reduced to $\leq$ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to $\leq$ 85% of RATED THERMAL POWER, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within $\pm$ 12 steps of the inoperable rod within the hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant From Small Ruptured Pipes Or From Cracks In Large Pipes Which Actuates The Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal At Full Power

Major Reactor Coolant System Pipe Rupture (Loss Of Coolant Accident)

Major Secondary System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

533 073

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown ~~control~~ and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

POSITION INDICATOR CHANNELS-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one rod postion indicator channel (excluding demand position indication) shall be OPERABLE for each shutdown control rod not fully inserted.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.

*With the reactor trip system breakers in the closed position.
#See Special Test Exception 3.10.5.

533 075

PART LENGTH ROD INSERTION LIMITS

DELETE

LIMITING CONDITION FOR OPERATION

3.1.3.7 All part length rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*

ACTION:

With a maximum of one part length rod not fully withdrawn, within one hour either:

a. Fully withdraw the rod, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.7 Each part length rod shall be determined to be fully withdrawn by:

a. Verifying the position of the part length rod prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER, and

b. Verifying, at least once per 31 days, that the breaker that supplies electric power to the drive mechanism is locked in the off position.

* See Special Test Exceptions 3.10.2 and 3.10.3.

533 076

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT		RESPONSE TIME
1.	Manual Reactor Trip	NOT APPLICABLE
2.	Power Range, Neutron Flux	$\leq$ 0.5 seconds*
3.	Power Range, Neutron Flux, High Positive Rate	NOT APPLICABLE
4.	Power Range, Neutron Flux, High Negative Rate	$\leq$ 0.5 seconds*
5.	Intermediate Range, Neutron Flux	NOT APPLICABLE
6.	Source Range, Neutron Flux	NOT APPLICABLE
7.	Overtemperature ΔT	$\leq$ 6.0 seconds*
8.	Overpower ΔT	NOT APPLICABLE
9.	Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10.	Pressurizer Pressure--High	$\leq$ 2.0 seconds
11.	Pressurizer Water Level--High	NOT APPLICABLE

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

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
OR, COINCIDENT WITH					
Steam Line Pressure-Low				1, 2, 3##	
Three Loops Operating	1 pressure/ loop	2 pressures any loops	1 pressure any 2 loops		14*
Two Loops Operating	1 pressure/ operating loop	1### pressure in any operating loop	1 pressure any operating loop		15
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2, 3	14*
6. AUXILIARY FEEDWATER PUMP START					
a. Steam Generator Water Level Low-Low	3/stm. gen.	2/stm. gen in any operating stm. gen.	2/stm. gen.	1, 2, 3, 4	14*
b. SI	See #1 above (All SI initiating functions and requirements)				
c. Station Blackout	2	2	2	1, 2, 3, 4	18
d. Main feed pump trip	2/pump	1/pump	1/pump	1, 2	17*
7 LOSS OF POWER					
a 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	3/Bus	2/Bus	2/Bus	1,2,3	19*
b 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1,2,3	19*

533 078

TABLE 3.3-3 (Continued)

ACTION 17 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour and the Minimum Channels OPERABLE requirement is met, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour.

b. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.1.

553 079

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
6. AUXILIARY FEEDWATER PUMP START		
a. Steam Generator Water Level Low-Low	$\geq$ 5% of narrow range instrument span each steam generator	$\geq$ 4% of narrow range instrument span each steam generator
b. S.I.	See 1 above (All S.I. Setpoints)	
c. Station Blackout	$\geq$ 57.5% Transfer Bus Voltage	$\geq$ 52.5% Transfer Bus Voltage
d. Trip of Main Feed Pump	N.A.	N.A.
7. LOSS OF POWER		
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2999 ± 60 volts with a 2.2 ± 0.03 second time delay	2912 ± 60 volts with a 3 ± 0.03 second time delay
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3744 ± 1.4 volts with a 60 ± 3 second time delay	3619 ± 1.4 volts with a 75 ± 3 second time delay

533 080

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
6. Steam Flow in Two Steam Lines-High Coincident with Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 13.0#/23.0##
b. Reactor Trip (from SI)	$\leq$ 3.0
c. Feedwater Isolation	$\leq$ 8.0
d. Containment Isolation-Phase "A"	$\leq$ 18.0#/28.0##
e. Auxiliary Feedwater Pumps	$\leq$ 60.0
f. Essential Service Water System	Not Applicable
g. Steam Line Isolation	$\leq$ 8.0
7. Containment Pressure--High-High	
a. Containment Quench Spray	$\leq$ 60.0
b. Containment Isolation-Phase "B"	$\leq$ 60.0
8. Containment Pressure-Intermediate High-High	
a. Steam Line Isolation	$\leq$ 7.0
9. Steam Generator Water Level Low-Low	
a. Auxiliary Feedwater Pumps	$\leq$ 60.0
10. Station Blackout	
a. Auxiliary Feedwater Pumps	Not Applicable
11. Main Feedwater Pump Trip	
a. Auxiliary Feedwater Pumps	Not Applicable
12. Steam Generator Water Level--High High	
a. Turbine Trip - Reactor Trip	$\leq$ 2.5
b. Feedwater Isolation	$\leq$ 11.0

See Attached Sheet

533 081

13. Loss of Power

a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≤ 13.3###
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≤ 11.5### with SI signal ≤ 64.0### with no SI signal

POOR ORIGINAL

TABLE 3.3-5 (Continued)

TABLE NOTATION

* Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps, and Low Head Safety Injection pumps.

\# Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\# Diesel generator starting and sequence loading delays included. Response time limit includes opening of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps.

\#\#\# The response times shown are based on the time from when the signal reaches the trip setting until the diesel generator is supplying the emergency bus.

533 083

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
4. STEAM LINE ISOLATION				
a. Manual	N.A.	N.A.	R	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3
c. Containment Pressure--Intermediate High-High	S	R	M	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low (T_{avg} -- Low) or Steam Line Pressure--Low	S	R	M	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION				
a. Steam Generator Water Level--High-High	S	R	M	1, 2, 3
6. AUXILIARY FEEDWATER PUMPS				
a. Steam Generator Water Level--Low-Low	S	R	M	1, 2, 3, 4
b. S.I.	See 1 above (all S.I. Surveillance Requirements)			
c. Station Blackout	N.A.	R	N.A.	1, 2, 3, 4
d. Main Feedwater Pump Trip	N.A.	N.A.	R	1, 2
7. LOSS OF POWER				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	M	1,2,3
b. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	M	1,2,3

533 084

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. STEAM LINE ISOLATION		
a. Manual	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Containment Pressure--Intermediate High-High	$\leq$ 20 psia	$\leq$ 22 psia
d. Steam Flow in Two Steam lines--High Coincident with T_{avg}--Low-Low Or Steam Line Pressure--Low	$\leq$ A function defined as follows: a Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load.	$\leq$ A function defined as follows: a Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 111.5% of full steam flow at full load.
	$T_{avg} \geq 543°F$	$T_{avg} \geq 541°F$
	$\geq$ 600 psig steam line pressure	$\geq$ 580 psig steam line pressure
5. TURBINE TRIP AND FEEDWATER ISOLATION		
a. Steam Generator Water level--High-High	$\leq$ 75% of narrow range instrument span each steam generator	$\leq$ 76% of narrow range instrument span each steam generator

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION, TURBINE TRIP AND FEEDWATER ISOLATION				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure-High	S	R	M	1, 2. 3
d. Pressurizer Pressure--Low Coincident with Pressurizer Water Level--Low	S	R	M	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	M	1, 2, 3
f. Steam Flow in Two Steam Lines--High Coincident with (T_{avg}--Low) or Steam Line Pressure--Low	S	R	M	1, 2, 3
2. CONTAINMENT SPRAY				
a. Manual Initiation	N.A.	N.A.	M(1)	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	M(2)	1, 2, 3, 4
c. Containment Pressure--High-High	S	R	M	1, 2, 3

T_{avg}--Low-Low

553 086

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 in that fire detection zone is required to be OPERABLE.

ACTION:

With one or more of the fire detection instrument(s) shown in Table 3.3-11 inoperable:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and 2

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 shall be demonstrated OPERABLE at least once per 6 months by performance of a CHANNEL FUNCTIONAL TEST.

4.3.3.7.2 The NFPA Code 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 between the local panels in Specification 4.3.3.7.2 and the control room shall be demonstrated OPERABLE at least once per 31 days.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period.

APPLICABILITY: At all times.

See attached sheet

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 fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for cont ed 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 psig, respectively, within the following 30 hours.

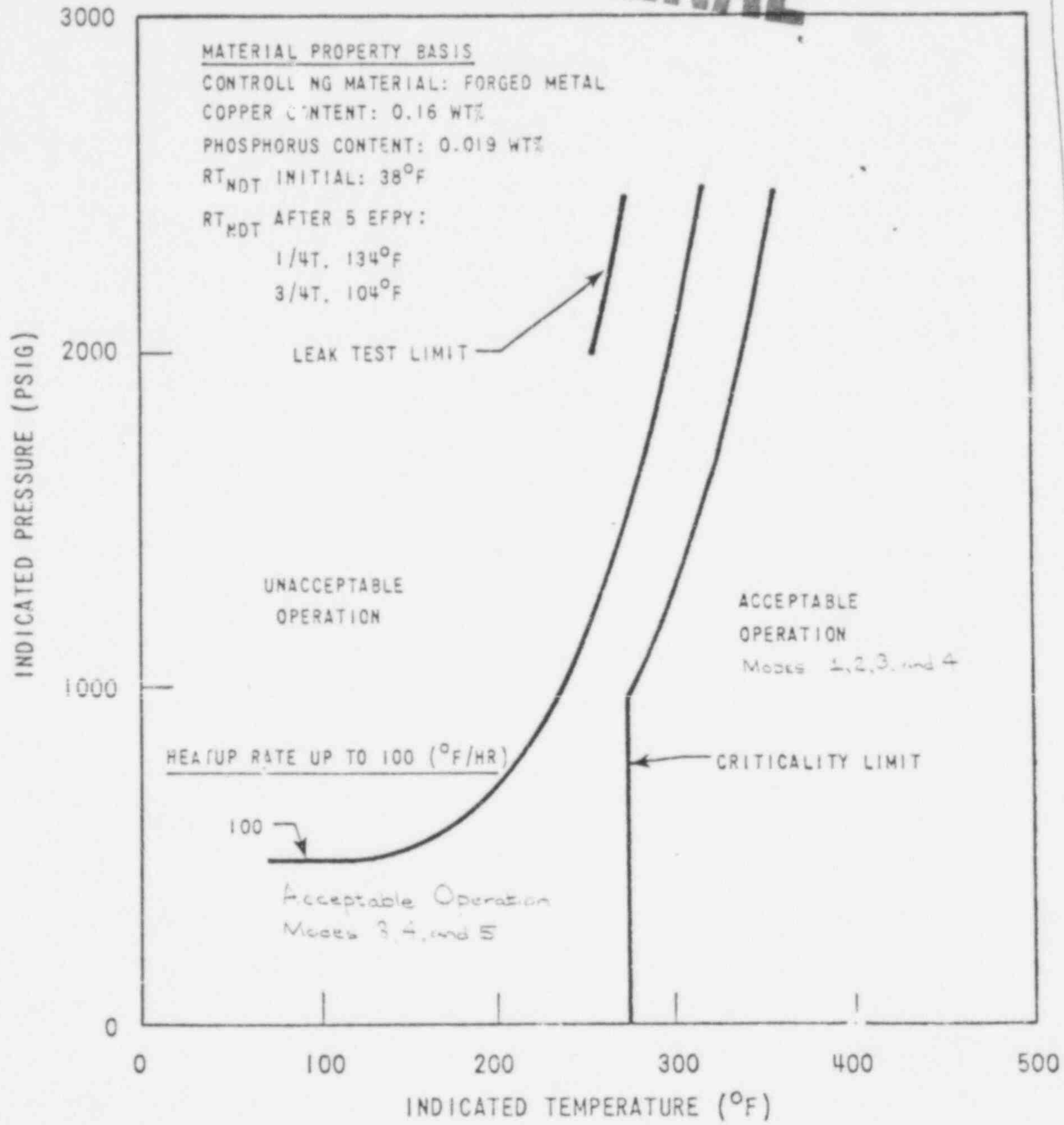
SURVEILLANCE REQUIREMENTS

4.4.9.1.1 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.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material prope ties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

533 088

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 lines.



Reactor Coolant System Temperature-Pressure Heatup Limitations

Figure 3.4-2 ~~North Anna Power Station No. 1~~ Reactor Coolant System ~~Heatup Limitations Applicable to 5 Effective Full Power Years and Contains Margins of 10°F and 60 PSIG for Possible Instrument Errors~~

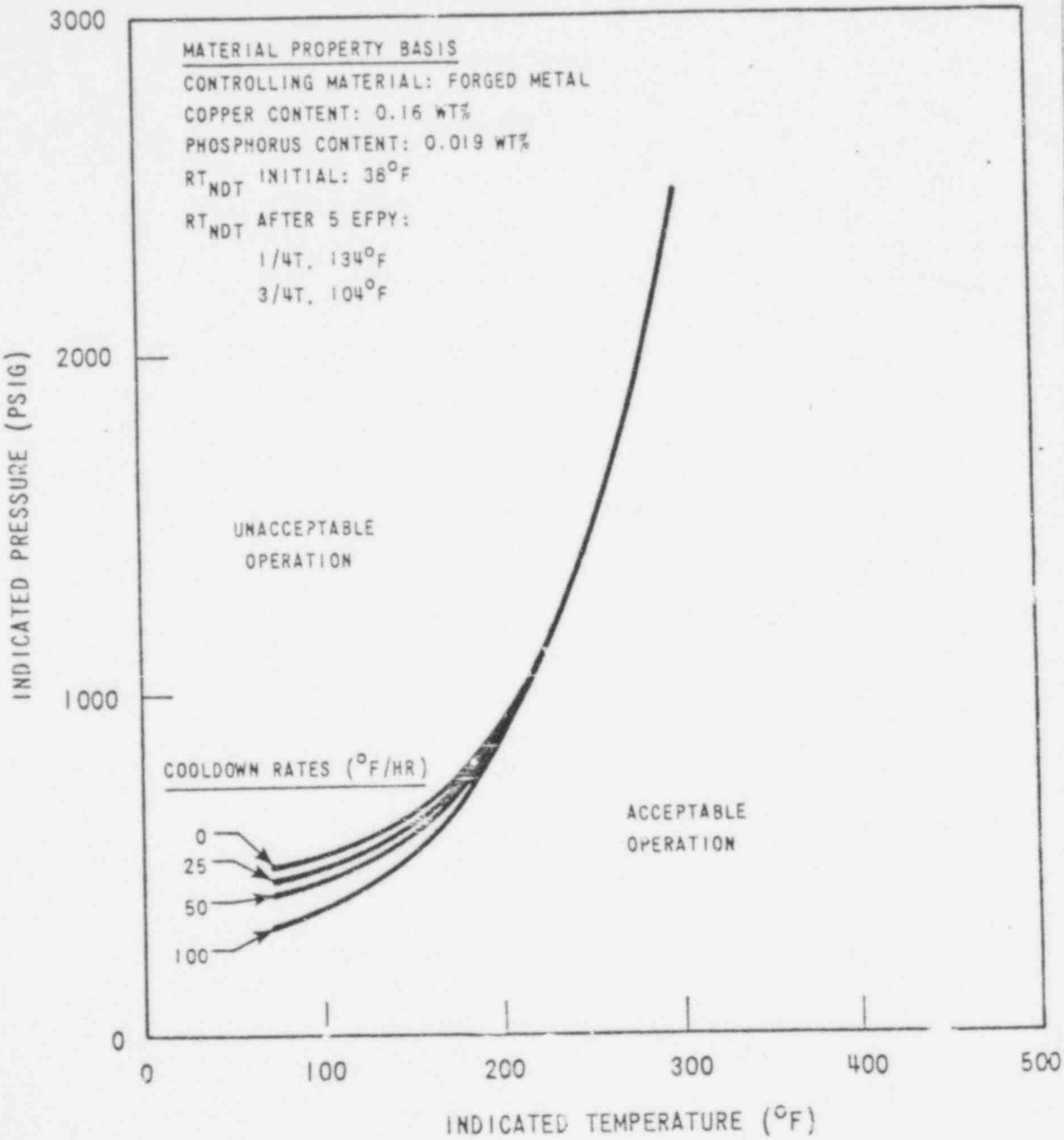


Figure 3.4-3 North Anna Power Station No. 1 Reactor Coolant System Cooldown Limitations ~~Applicable to 5 Effective Full Power Years and Contains Margins of 10°F and 60 PSIG for Possible Instrument Errors~~

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

CAPSULE	REMOVAL INTERVAL
V	1st Refueling
X	10 years
W	10 years (Reinsert in Location V)
Y	10 years (Reinsert in Location X)
W	20 years
Z	20 years (Reinsert in Location V)
Y	30 years
U	30 years (Reinsert in Location X)
T	(Extra)
S	(Extra)

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. Two power operated relief valves (PORVs) with a lift setting of less than or equal to ~~475~~ 505 psig*, or

b. A reactor coolant system vent of greater than or equal to 2.07 square inches, or

c. A maximum pressurizer water volume of 457 cu. ft.#

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to ~~340°F~~ 320°F, except when the reactor vessel head is unbolted.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 2.07 square inch vent(s) within the next 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

b. With both PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

*Less than or equal to ~~405~~ 430 psig with an RCS cold leg temperature less than 140°F.

#Only applicable for RCS cold leg temperatures ≥ ~~320°F~~ 300°F.

533 093

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. Testing pursuant to Specification 4.0.5.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

---

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify ~~these valves open at least once per 31 days.~~ the valve(s) open at least once per 31 days when being used for overpressure protection.

533 094

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. An OPERABLE flow path capable of transferring fluid to the Reactor Coolant System when taking suction from the refueling water storage tank on a safety injection signal or from the containment sump when suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump. 2

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumlated actuation cycles to date.

c. The provisions of Specification 3.0.4 are not applicable to 3.5.2.a and 3.5.2.b for one hour following heatup above 320°F or prior to cooldown below 320°F.

533 095

ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE low head safety injection pump,# and

c. An OPERABLE flow path capable of transferring fluid to the reactor coolant system when taking suction from the refueling water storage tank upon being manually realigned or from the containment sump when the suction is transferred during the recirculation phase of operation or from the discharge of the outside recirculation spray pump. 2

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of the low head safety injection pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

See Attached Sheet

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

See Attached Sheet

553 096

A maximum of one centrifugal charging pump and one low head safety injection pump shall be operable whenever the temperature of one or more of the RCS cold legs is less than or equal to 320°F.

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is 320°F by verifying that their switches in the Control Room are in the pull to lock position.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen analyzers (shared with Unit 2) shall be OPERABLE.

APPLICABILITY: MODES ˙ and 2.

ACTION:

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

SURVEILLLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent (± .25%) hydrogen, balance nitrogen, and

b. Four ~~Volume~~ volume percent (± .25%) hydrogen, balance nitrogen.

533 098

WASTE GAS CHARCOAL FILTER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.4.3 A waste gas charcoal filter system (shared with Unit 2) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the waste gas charcoal filter system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 The waste gas charcoal filter system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Initiating flow through the HEPA filter and charcoal adsorber train using the process vent blowers and verifying that the purge system operates for at least 15 minutes,

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

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a., C.5.c and C.5.d of Regulatory Guide 1.52, ~~Revision 1, July 1976~~ Revision 2, March 1978, and the system flow rate is 300 cfm ± 10% (except as shown in Specifications 4.6.4.3.e and 4.6.4.3.f).

533 099

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b. of Regulatory Guide 1.52, ~~Revision 1, July 1976~~, meets the laboratory testing criteria of Regulatory Position C.6.a. of Regulatory Guide 1.52, ~~Revision 1, July 1976~~. Revision 2, March 1978

3. Verifying a system flow rate of 300 cfm ±10% during system operation when tested in accordance with ANSI N 510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, ~~Revision 1~~, Revision 2 March 1978 ~~July 1976~~, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, ~~Revision 1, July 1976~~. Revision 2, March 1978

d. At least once per 18 months by:

1. Verifying that the pressure drop across the HEPA filter and charcoal adsorber assembly is $\leq$ 8.5 inches Water Gauge while operating the filter train at a flow rate of 300 cfm ± 10%.
less than

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 300 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 300 cfm ± 10%.

PLANT SYSTEMS

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when the water level of the North Anna Reservoir exceeds 256 feet Mean Sea Level USGS datum, at the main reservoir spillway.

APPLICABILITY: At all times.

ACTION:

With the water level at the main reservoir spillway above elevation 256 feet Mean Sea Level USGS datum:

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

b. Initiate and complete within 36 hours, the following flood protection measures:

   1. Stop the circulating water pumps, and

   2. Close the condenser isolation valves

SURVEILLANCE REQUIREMENTS

4.7.6.1 The water level at the main reservoir spillway shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation 255 feet Mean Sea Level USGS datum.

b. Measurement at least once per 2 hours when the water level is equal to or above 255 feet Mean Sea Level USGS datum.

533 101

3/4.7.11 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.11.1 Each sealed source containing radioactive material either in excess of ~~those quantities of byproduct material listed in 10 CFR 30.71 or > 0.1 micracuires of any other material, including alpha emitters, shall be free of > 0.005 microcuries of removable contamination.~~ See Attached Sheet

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limits shall be immediately withdrawn from use and:

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

4.7.11.1.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per six months for all sealed sources containing radioactive materials.

100 microcuries of beta and/or gamma emitting
material or 5 microcuries of alpha emitting
material should be free of greater than or equal
to

POOR ORIGINAL

SURVEILLANCE REQUIREMENTS (Continued)

1. With a half-life greater than 30 days (excluding Hydrogen 3), and

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

4.7.11.1.3 Reports - A Special Report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

LOW PRESSURE CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.14.2 The following low pressure CO_2 systems shall be OPERABLE with a minimum of ~~10 and 6~~ 3.5 tons in the storage tanks at a minimum pressure of 275 psig.

a. Cable tunnels and vaults
b. Charcoal filters
c. Emergency diesel generator rooms

APPLICABILITY: Whenever equipment in the low pressure CO_2 protected areas is required to be OPERABLE.

ACTION:

a. With one or more of the above required low pressure CO_2 systems inoperable, establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s) within 1 hour; 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 pursuan 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. 2

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

SURVEILLANCE REQUIREMENTS

4.7.14.2 Each of the above required low pressure CO_2 systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying CO_2 storage tank level and pressure, and

b. At least once per 18 months by verifying:

1. The system valves and associated ventilation dampers actuate manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

533 105

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) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

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. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provides assurance of fuel rod integrity during continued operation. In addition those accident analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

~~The restriction prohibiting part length rod insertion ensures that adverse power shapes and rapid local power changes which may effect DNB considerations do not occur as a result of part length rod insertion during operation.~~

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 500°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

533 106

TABLE B 3/4 4-1

REACTOR VESSEL TOUGHNESS TABLE (UNIT 1)

Component	Heat No.	Material Type	Cu (%)	P (%)	NDTT (°F)	Minimum 50 ft-lb/35 mil Temp. (°F) Parallel to Major Working Direction	Minimum 50 ft-lb/35 mil Temp. (°F) Normal to Major Working Direction	RT_{NDT} (°F)	Average Upper Shelf (ft-lb) Parallel to Major Working Direction	Average Upper Shelf (ft-lb) Normal to Major Working Direction
Cl. Hd. Dome	53565-3	A533,B,Cl.1			-31	14	34*	-26	87(a)	
Cl. Hd. Flange	4984	A508,Cl.2			-40	<-76	-56*	-40	141(a)	
Ves. Flange	4982	A508,Cl.2			-22	<-70	-50*	-22	161(a)	
Inlet Nozzle	4964	A508,Cl.2			-31	14	34*	-26	108(a)	
Inlet Nozzle	4966	A508,Cl.2			-22	10	30*	-22	88(a)	
Inlet Nozzle	4968	A508,Cl.2			-22	43	63*	3	80(a)	
Outlet Nozzle	4963	A508,Cl.2			-13	43	63*	-3	100(a)	
Outlet Nozzle	4965	A508,Cl.2			-22	14	34*	-22	90(a)	
Outlet Nozzle	4967	A508,Cl.2			-4	34	54*	-4	90(a)	
Upper Shell	4952	A508,Cl.2			2	46	66*	6	60(a)	
Inter. Shell	4958	A508,Cl.2	0.12	0.009	-31	20	77(s)	17(s)	95(a)	92(s)
Lower Shell	4979	A508,Cl.2	0.16	0.019	-13	40	98(s)	38(s)	74(a)	85(s)
Bot. Hd. Seg.	53647-3	A533,B,Cl.1			-31	27	47*	-13	65.5(a)	
Bot. Hd. Seg.	53648-4	A533,B,Cl.1			-13	27	47*	-13	77(a)	
Bot. Hd. Dome	53774	A533,B,Cl.1			-22	32	52*	-8	67(a)	
Weld		Weld	~~0.06~~ 0.086	0.020	-13		79	19		102
Haz		Haz			-22		39	-21		142

*Estimated temperature ~~based on Regulatory Review Plan, Branch Technical Position MTEB-5-2~~ per NRC Standard 5.3.2

(a) Minimum energy at highest test temperature (≤ 68°F) - % ~~shoer~~ shear not reported

(s) Average tranverse data obtained from surveillance program.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 2.07 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 is $\leq$ ~~340~~ 320°F, and the Reactor Vessel Head is bolted. Either PORV 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 $\leq$ 50°F above the RCS cold leg temperature, or (2) the start of a charging pump and its injection into a water solid RCS. When the Reactor Vessel Head is unbolted, a RCS pressure of less than 100 psig will lift the head, thereby creating a relieving capability equivalent to at least one PORV.

When the temperature of the RCS cold legs is between ~~320~~ 300°F and ~~340~~ 320°F, overpressure protection can also be provided by a bubble in the pressurizer. In such a case, a maximum pressurizer water volume of 457 cu. ft. has been selected to provide at least 10 minutes for operator response in the event of a malfunction resulting in maximum flow from one charging pump.

insert on page B 3/4 4-11

3/4.7.11 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Leakage of sources excluded from the requirements of this specification represent less than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested. See Attached Sheet

3/4.7.12 SETTLEMENT OF CLASS 1 STRUCTURES

In order to assure that settlement does not exceed predicted and allowable settlement values, a program has been established to conduct a survey of a specified number of points at the site on a semi-annual basis. The first survey was conducted in May 1976 to establish base-line elevations for most of the points. Where applicable, the base-line elevations of points established subsequent to the May 1976 survey have been adjusted to the May 1976 survey by an evaluation of the settlement records of settlement points on the same structure or on nearby structures. Baseline elevations for some points were established on dates other than May 1976 as indicated in Table 3.7-5. Additional surveys will be performed semiannually.[2] The determination of the elevation of the points shall be by precise leveling with second order Class II accuracy as defined by the U. S. Department of Commerce, National Oceanic and Atmospheric Administration, National Ocean Survey, 1974. When any settlement point listed in Table 3.7-5 is inaccessible during a survey, comparison to allowable settlement shall be based on settlement of other points on the same structure or on nearby structures having similar foundation conditions. When any settlement point listed in Table 3.7-5 is dislocated or replaced, a documented review of the settlement records of points on the same structure and additionally points on nearby structures having similar foundation conditions shall provide a new reference elevation for the point that reflects the estimated settlement since the base-line survey. If the total settlement or differential settlement exceeds 75 percent of the allowable value, the frequency of surveillance shall be increased as dictated by the engineering review. 8

Allowable differential movement is controlled by pipe deflections permitted by fixation points in buildings. The items limiting differential settlement are as follows:

533 109

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

POOR ORIGINAL

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material.

The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The balance of the void length in the part length rods shall contain aluminum oxide.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

533 111

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

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

6.2 ORGANIZATION

OFFSITE

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

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

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

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

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

f. A Fire Brigade of at least 5 members shall be maintained onsite at all times.# The Fire Brigade shall not include the minimum shift crew shown in Table 6.2-1 or any personnel required for other essential functions during a fire emergency.

See Attached Sheet

533 112

* Fire Brigade composition may be less than the minimum requirement for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirement.

POOR ORIGINAL

REVIEW (Cont'd)

Specifications Technical

1. Violations of applicable codes, regulations, orders, ~~technical specifications~~, license requirements or internal procedures or instructions having safety significance;

2. Significant operating abnormalities or deviations form normal or expected performance of station safety-related structures, systems, or components; and

Technical

3. Reportable occurrences as defined in the station ~~technical specifications~~.

Specifications

Review of events covered under this paragraph shall include the results of any investigations made and recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event.

e. Any other matter involving safe operation of the nuclear power stations which a duly appointed subcommittee or committee member deems appropriate for consideration, or which is referred to the SyNSOC by the Station Nuclear Safety and Operating Committee.

AUDITS

6.5.2.8 Audits of station activities shall be performed under the cognizance of the SyNSOC. These audits shall encompass:

a. The conformance of facility 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 facility staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in facility 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 Station Emergency Plan and implementing procedures at least once per 24 months.

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

6.8 PROCEDURES

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

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, ~~November, 1972.~~ Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e.. Emergency Plan implementation.

f. Fire Protection Program Implementation.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the SNSOC and approved by the Station Manager prior to implementation and 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 supervisory staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the SNSOC and approved by the Station Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

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

greater than 100 mrem/hr but less than 1000 mrem/hr

6.12 HIGH RADIATION AREA

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

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

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

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Plant Health Physicist.

*Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

3/4.7.7 CONTROL ROOM EMERGENCY HABITABILITY SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.7.1 The following control room emergency habitability systems shall be OPERABLE:

a. The emergency ventilation system.

b. The bottled air pressurization system.

c. Two air conditioning systems.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: See Attached Sheet

~~With either the emergency ventilation system or the bottled air pressurization system and one air conditioning system inoperable, restore the inoperable systems 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.~~

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each control room emergency ventilation system shall be demonstrated OPERABLE:

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

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

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 1000 cfm ± 10% (except as shown in Specifications 4.7.7.1.e and 4.7.7.1.f).

553 117

ACTION:

a. With either the emergency ventilation system or the bottled air pressurization 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 at least COLD SHUTDOWN within the following 30 hours.

b. With both the emergency ventilation system and the bottled air pressurization system inoperable, restore at least one of these systems to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

c. With one air conditioning system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANBY within the next 6 hours or be in at least COLD SHUTDOWN within the following 30 hours.

and

d. With both air conditioning systems inoperable, restore at least one sys to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in at least COLD SHUTDOWN within the following 30 hours.

POOR ORIGINAL

3/4.7.8 SAFEGUARDS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two safeguards area ventilation systems (SAVS) shall be OPERABLE with:

a. one SAVS exhaust fan

b. one auxiliary building HEPA filter and charcoal adsorber assembly (shared with Unit 2)

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one SAVS 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.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each SAVS system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Initiating, from the control room, flow through the auxiliary building HEPA filter and charcoal adsorber assembly and verifying that the SAVS operates for at least 10 hours with the heater on.

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

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 6,300 cfm ± 10%. (except as shown in Specification 4.7.8.1.e and 4.7.8.1.f).

533 119

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel

APPLICABILITY: During CORE ALTERATIONS while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all CORE ALTERATIONS. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during CORE ALTERATIONS.

3/4.10 SPECIAL TEST EXCEPTIONS

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 control rod worth and SHUTDOWN MARGIN provided.

a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s). ~~and~~

~~b. All part length rods are withdrawn to at least the 180 step position and OPERABLE.~~

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion or the part length rods not within their withdrawal limits, initiate and continue boration at ≥ 10 gpm of at least 20,000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 10 gpm of at least 20,000 ppm boric acid solution 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~~ rod either partially or fully withdrawn shall be determined at least once per 2 hours.

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

~~4.10.1.3 The part length rods that are partially withdrawn, shall be demonstrated OPERABLE by moving each part length rod ≥ 10 steps within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1~~.

533 121

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.1.3.7~~, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained $\leq$ 85% of RATED THERMAL POWER, and

b. The limits of Spec fications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, ~~3.1.3.7~~, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWE: sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be < 85% of RATED THERMA' POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at the following frequencies during PHYSICS TESTS:

a. Specification 4.2.2 - At least once per 12 hours.

b. Specification 4.2.3 - At least once per 12 hours.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6, ~~and 3.1.3.7~~ may be suspended during the performance of PHYSICS TESTS provided:

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

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at ≤ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

533 123