ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION

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CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) 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 3.6.4.

b. The equipment hatch is closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

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

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

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/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" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$ - 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 (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

[illegible]

TABLE 2.2-1 (Continued)

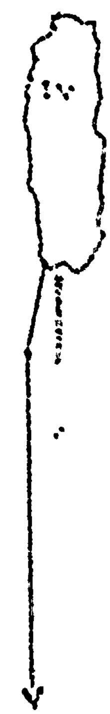
TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00068/°F for T > T" ~~and K_6~~ = 0 for T ≤ T"
T	=	As defined in Note 1,
T"	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 574.2°F),
S	=	As defined in Note 1, and
f (ΔI)	=	As defined in Note 1.

NOTE 4: (This note number is not used.)

If no allowable value is specified as indicated by [ ], the trip set point shall also be the allowable value.



REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 The group step counter demand position indicator shall be OPERABLE and capable of determining within ± 2 steps the demand position for each shutdown and control rod not fully inserted.

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

ACTION:

With less than the above required group step counter demand position indicator(s) OPERABLE, open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3.1 Each of the above required group step counter demand position indicator(s) shall be determined to be OPERABLE by movement of the associated control rod at least 10 steps in any one direction at least once per 31 days.

4.1.3.3.2 ~~A CHANNEL CHECK CALIBRATION AND ANALOG CHANNEL OPERATIONAL TEST shall be performed per Table 4.1-1~~.

INSERT

OPERABILITY of the group step counter demand position indicator shall be verified in accordance with Table 4.1-1.

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.~~4~~ 5

INSERT

OPERABILITY of the group step counter demand position indicator shall be verified in accordance with Table 4.1-1.

5

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f. Steam Line Flow--High Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Steam Generator Pressure--Low	1/steam generator	1/steam ~~line~~ generator in any two steam ~~lines~~ generators	1/steam generator in any two steam ~~lines~~ generators	1, 2, 3*	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	15
2. Containment Spray					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15
Containment Pressure--High	3	2	2	1, 2, 3	15
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam ~~lines~~ generators	1/steam generator in any two steam ~~lines~~ generators	1, 2, 3	15
or T_{avg}--Low	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	15
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater###					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWANCE (TA)	Z	S	TRIP SETPOINT	ALLOWABLE VALUE#
4. Steam Line Isolation (Continued)					
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-High Coincident with:	[ ]	[ ]	[ ]	≤30.0 psig	≤[ ] psig
Containment Pressure--High	[ ]	[ ]	[ ]	≤6.0 psig	≤[ ] psig
~~e.~~ d. Steam Line Flow--High	[ ]	[ ]	[ ]	≤A function defined as follows: A Δp corresponding to 0.64 x 10^6 lbs/hr at 0% load increa-ing linearly to a Δp corresponding to 3.84 x 10^6 lbs/hr at full load.	[ ]
Coincident with: Steam Line Pressure--Low	[ ]	[ ]	[ ]	≥600 psig	≥[ ] psig
or T_{avg}--Low	4.0	2.0	1.0	≥543°F	≥542.5°F
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A.	N.A.	N.A.
b. Safety Injection	see item 1			See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.	

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High (Particulate or Gaseous (See Note 1.))	1	1*	All*	Particulate $<6.1\times10^5$ CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 for MODES 5 AND 6
b. RCS Leakage Detection Particulate Radioactivity or Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	26
2. Spent Fuel Storage Pool Areas					
a. Unit 3 Radioactivity - High Gaseous	1	1	**	$<5.5\times10^{-2}$ $\frac{\mu Ci}{cc}$	28
b. Unit 4 Radioactivity-High Gaseous#	1	1	**	$<2.8\times10^{-2}$ $\frac{\mu Ci}{cc}$ (SPING) or $<1.0\times10^6$ CPM (PRMS)	28

TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTIONS
14. In Core Thermocouples (Core Exit Thermocouples)	4/core quadrant	2/core quadrant	1, 2, 3	31, 32
15. Containment High Range Area Radiation	2	1	1, 2, 3	34
16. Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38
17. Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32
18. Containment Hydrogen Monitors	2	1	1, 2	35
19. High Range-Noble Gas Effluent Monitors				
a. Plant Vent Exhaust	1	1	ALL	34
b. Unit 3-Spent Fuel Pit Exhaust	1	1	ALL	34
c. Condenser Air Ejectors	1	1	1, 2, 3	34
d. Main Steam Lines	1	1	1, 2, 3	34
20. RWST Water Level	2	1	1, 2, 3	31, 32
21. Steam Generator Water Level (Narrow Range)	2/stm. gen.	1/stm. gen.	1, 2, 3	31, 32
22. Containment Isolation Valve Position Indication*	1/valve	1/valve	1, 2, 3	39

TABLE NOTATIONS

1. A channel is eight sensors in a probe. A channel is OPERABLE if a minimum of four sensors are OPERABLE.
2. Inputs to this instrument are from instrument items 3, 4, 5 and 14 of this Table.

---

*Applicable for containment isolation valve position indication designated as post-accident monitoring instrumentation (containment isolation valves which receive containment isolation Phase A, Phase B, or containment ventilation isolation signals).

Quadrant

TABLE 3.3-6

FIRE DETECTION INSTRUMENTS FOR ESSENTIAL EQUIPMENT

ADD

INSTRUMENT LOCATION FIRE ZONE AREA	TOTAL NUMBER OF INSTRUMENTS HEAT (x/y)*	FLAME (x/y)*	SMOKE (x/y)*
4 - Aux. Bldg. Corridor E. 10'			(2/0)
5 - Chem. Drain/Laundry/Shower Tank Room			(2/0)
9 - Laundry/Chemical Drain Tank Room			(1/0)
10 - Pipeway			(11/0)
11 - Unit 3 RHR Heat Exchanger Room			(5/0)
12 - RHR Pump 3A Room			(2/0)
13 - RHR Pump 3B Room			(2/0)
14 - Unit 4 RHR Heat Exchanger Room			(5/0)
15 - RHR Pump 4A Room			(2/0)
16 - RHR Pump 4B Room			(2/0)
19 - Unit 3 W Elect Penet Room			(5/0)
20 - Unit 3 S Elect Penet Room			(11/0)
21 - Instrument Shop			(2/0)
22 - Radioactive Laboratory			(2/0)
25 - Aux. Bldg. Elect. Equipmt. Room	(2/0)		(5/0)
26 - Unit 4 N Elect Penet Room			(8/0)
27 - Unit 4 W Elect Penet Room			(6/0)
30 - Unit 4 Piping and Valve Room			(4/0)
40 - Unit 3 Piping and Valve Room			(4/0)
45 - Unit 4 Charging Pump Room	(0/4)		(3/0)
47 - Unit 4 Component Cooling Water Area	(0/4)	(5/2)	
54 - Unit 3 Component Cooling Water Area	(0/4)	(4/2)	
55 - Unit 3 Charging Pump Room	(0/4)		(3/0)
58 - Aux Bldg Corridor, El. 18'			(18/0)
59 - Unit 4 Containment Electrical Penet. Area**			(10/0)
60 - Unit 3 Containment Electrical Penet. Area**			(16/0)
61 - Reactor Control Rod Eqpmt Room - Unit 4			(4/0)
62 - Computer Room			(11/0)
63 - Reactor Control Rod Eqpmt Room - Unit 3			(4/0)
67 - 4160V Switchgear 4B			(10/0)
68 - 4160V Switchgear 4A			(6/0)
70 - 4160V Switchgear 3B			(10/0)
71 - 4160V Switchgear 3A			(6/0)
72 - Diesel Generator 3B	(0/3)	(1/0)	(1/0)
73 - Diesel Generator 3A	(0/3)	(1/0)	(1/0)
74 - Day Tank Room 3B	(1/1)		
75 - Day Tank Room 3A	(1/1)		
76 - Unit 4 Turbine Lube Oil Reservoir	(1/0)		
79A- North-South Breezeway	(0/6)		(4/0)
81 - Unit 4 Main Transformer	(1/0)		
82 - Unit 4 Aux Transformer Area	(1/0)		
84 - Unit 3 and 4 Aux Feedwater Pump Area (DC Enclosure Bldg.)			(3/0)

TABLE 3.3-6 (Continued)

FIRE DETECTION INSTRUMENTS FOR ESSENTIAL EQUIPMENT

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS		
FIRE ZONE AREA	HEAT (x/y)*	FLAME (x/y)*	SMOKE (x/y)*
		ADD	
87 - Unit 3 Aux Transformer Area	(1/0)		
93 - 480V Load Center 4A and 4B			(1/0)
94 - 480V Load Center 4C and 4D			(2/0)
95 - 480V Load Center 3A and 3B			(1/0)
96 - 480V Load Center 3C and 3D			(2/0)
97 - Mechanical Equipment Room			(1/0)
98 - Cable Spreading Room			(16/15)
101- RPI Inverter and MG Sets			(1/0)
102- Battery Rack 4B	(1/0)		
103- Battery Rack 3A	(1/0)		
104- RPI Inverter and MG Sets			(2/0)
106- Control Room	(1/0)		(17/0)
108A- Train A Inverters			(3/4)
108B- Train B Inverters			(4/4)
109- Battery Rack 4A	(1/0)		
110- Battery Rack 3B	(1/0)		
113- Unit 4 Feedwater Platform		(2/0)	
116- Unit 3 Feedwater Platform		(2/0)	
119- Unit 4 Intake Cooling Water Pump Area		(4/0)	
120- Unit 3 Intake Cooling Water Pump Area		(4/0)	
132- Control Room Electrical Chase			(1/0)
133- Diesel Generator 4B	(5/5)	(3/0)	(5/0)
134- 4160V Switchgear 3D Room			(2/0)
135- Diesel Generator 4B Control Panel Room			(2/0)
136- Diesel Generator 4B Fuel Transfer Pump			(2/0)
138- Diesel Generator 4A	(5/5)	(3/0)	(5/0)
139- 4160V Switchgear 4D Room			(2/0)
140- Diesel Generator 4A Control Panel Room			(2/0)
141- Diesel Generator 4A Fuel Transfer Pump			(2/0)
N/A - 18' level of the Turbine Area	(N/A)#	(N/A)#	(N/A)#

TABLE NOTATIONS

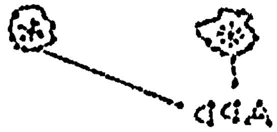
CENTER *

~~(x/y)~~: x is number of Function A (early warning fire dectection and notification only) instruments.
y is number of Function B (actuation of Fire Suppression Systems and early warning fire detection and notification) instruments.

DELETE

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

A fire watch patrol shall be established to inspect the 18 foot level of the Turbine Area once each hour.



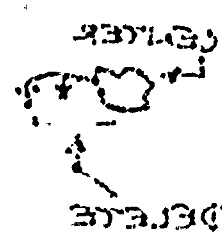


TABLE 4.3-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluents Line	D	P	R(2)*	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(2)	Q(1)
2. Flow Rate Measurment Devices			DELETE	
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R*	Q
b. Steam Generator Blowdown Effluent Lines	D(3)	N.A.	R	Q

~~*Channel calibration frequency shall be at least once per 18 months.~~

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates ~~measures~~ measured levels above the Alarm/Trip Setpoint.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the ~~National Bureau of Standards (NBS)~~ National Institute of Standards and Technology (NIST) or using standards that have been obtained from suppliers that participate in measurement assurance activities with ~~NBS~~ NIST. 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.

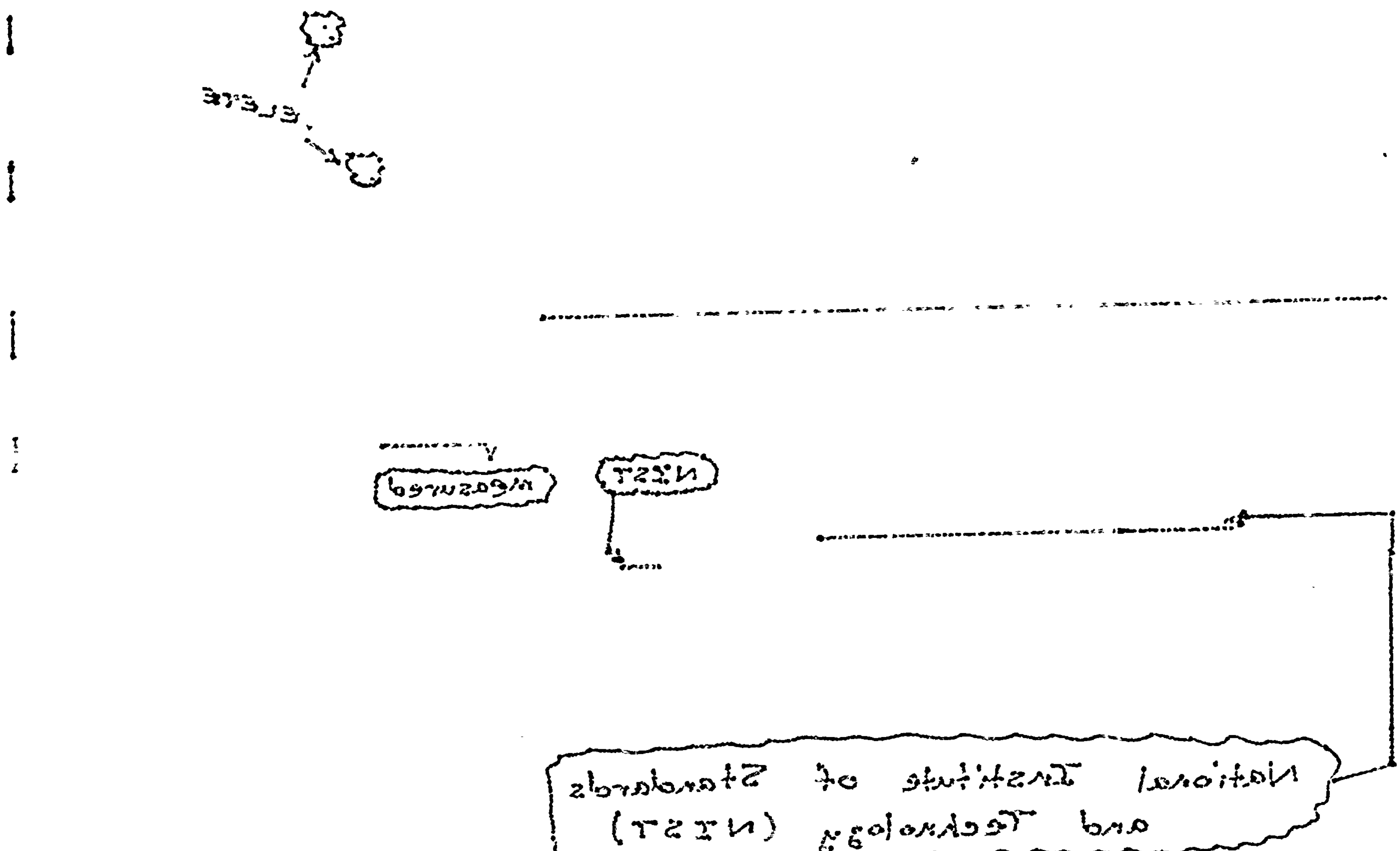

National Institute of Standards
and Technology (NIST)
NIST
measured

TABLE 3.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Effluent System Flow Rate Measuring Device	1	*	46
e. Sampler Flow Rate Measuring Device	1	*	46
5. Unit 3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Sampler Flow Rate Measuring Device	1	*	46

TABLE 4.3-[illegible] (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Condenser Air Ejector Vent System (Continued)					
e. Sample Flow Rate Measuring Device	D	N.A.	R	N.A.	##
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	R (3,~~6~~)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	D	N.A.	R ~~(6)~~	N.A.	*
e. Sampler Flow Rate Measuring Device	D	N.A.	R ~~(6)~~	N.A.	*

DELETE "(6)"

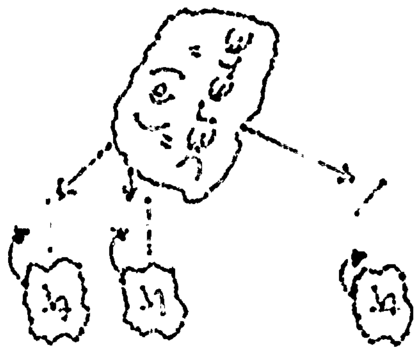


TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Unit 3 Spent Fuel Pit Building Vent					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Measuring Device	D	N.A.	R	N.A.	*

TABLE NOTATION

* At all times.
** During GAS DECAY TANK SYSTEM operation.
\# Applies during MODE 1, 2, 3 and 4.
\## Applies during MODE 1, 2, 3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that if the instrument indicates measured levels above the Alarm Setpoint, alarm annunciation occurs in the control room (for PRMS only) and in the computer room (for SPING only).

NIST

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

NATIONAL INSTITUTE OF STANDARDS AND TECHNOLOGY (NIST)

TABLE 4.3-6 (Continued)

TABLE NOTATIONS (Continued)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal.

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

~~(6) CHANNEL CALIBRATION frequency shall be at least once per 18 months.~~

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER		FUNCTION
Unit 3	Unit 4	High-Head Safety Injection Check Valves
3-874A	4-874A	Loop A, hot leg
3-875A	4-875A	cold leg
3-873A	4-873A	cold leg
3-874B	4-874B	Loop B, hot leg
3-875B	4-875B	cold leg
3-873B	4-873B	cold leg
3-875C	4-875C	Loop C, cold leg
3-873C	4-873C	cold leg
		Residual Heat Removal Line Check Valves
3-876A ~~3-876-A~~	4-876A	Loop A, cold leg
	4-876E	
3-876B	4-876B	Loop B, cold leg
3-876D	4-876D	
3-876C	4-876C	Loop C, cold leg
3-876E		
MOV3-750	MOV4-750	Loop A, hot leg to RHR
MOV3-751	MOV4-751	Loop ~~A~~ C, hot leg to RHR

ACCEPTABLE LEAKAGE LIMITS

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable provided that the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between previously measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

AND

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT ~~ANY~~ ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
4. Radiochemical Isotopic Determination Including Gaseous Activity	Monthly	1, 2, 3, 4
5. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
6. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

AND

SURVEILLANCE REQUIREMENTS (Continued)

d. Assuring that the observed lift-off force for each tendon exceeds the minimum required lift-off force. Required lift-off forces shall be calculated individually for each surveillance tendon prior to the beginning of each surveillance, and should consider such factors as:

   1) Prestressing history;
   2) Friction losses; and
   3) Time-dependent losses (creep, shrinkage, relaxation), considering time elapsed from prestressing.

e. Verifying the OPERABILITY of the sheathing filler grease by:

   1) Minimum grease coverage exists for the different parts of the anchorage system, and

   2) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through visual inspection that no unacceptable levels of corrosion exist on the end anchorages and no unacceptable cracking exists in the concrete adjacent to the end anchorages. Determination of acceptance levels shall be by engineering evaluation of the areas in question. If unacceptable conditions are found, the tendons inspected during the previous surveillance shall be examined to determine whether the corrosion levels or concrete cracking have increased since the previous surveillance. Inspection of adjacent concrete surfaces shall be performed concurrently with the containment tendon surveillance (Technical Specification 4.6.1.6.1).

4.6.1.6.3 Containment Surfaces. In accordance with 10 CFR 50, Appendix J. Section V. A, a visual inspection of the accessible interior and exterior surfaces of the containment, including the liner plate, shall be performed during the shutdown for (but prior to) each Type A containment leakage rate test (Technical Specification 4.6.1.2~~.1~~). The purpose of this inspection shall be to identify any evidence of structural deterioration which may affect containment structural integrity or leaktightness. The visual inspection shall be general in nature; its intent shall be to detect gross areas of widespread cracking, spalling, gouging, rust, weld degradation, or grease leakage. The visual examination may include the utilization of binoculars or other optical devices. Corrective actions taken, and recording of structural deterioration and corrective actions, shall be in accordance with 10 CFR 50, Appendix J, Section V. A. Records of previous inspections shall be reviewed to verify no apparent changes in appearance. The first inspection performed will form the baseline for future surveillances.

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Three emergency containment filtering units shall be OPERABLE.

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

ACTION:

With one emergency containment filtering unit inoperable, restore the inoperable filter 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

4.6.3 Each emergency containment filtering unit shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release or (3) after every 720 hours of system operation by:

1) Performance of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP and halogenated hydrocarbons at the system flow rate of 37,500 cfm ±10%;

2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portions of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and performed in accordance with ANSI N-510-1975, meets the acceptance criteria of greater than 99.9% removal of elemental iodine; and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the criteria of position ~~C.6a~~ of Regulatory Guide 1.52, Rev. 2; and
should read C.6.a

3) Verifying a system flow rate of 37,500 cfm ±10% and a pressure drop across the HEPA and charcoal filters of less than 6 inches water gauge during system operation when tested in accordance with ANSI N510-1975;

should read C.6.3

PLANT SYSTEMS

3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, fire barrier penetration seals, 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, piping, and ventilation duct penetration seals) shall be OPERABLE. (INSERT)

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type 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 will be inspected every 15 years.

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained greater than or equal to elevation 56' - 10"~~*~~ the spent fuel storage pool.~~**~~ (*)

DELETE

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

~~*During spent fuel rerack operation, the water level may be lowered to a level justified by an engineering safety evaluation. There will be no movement of fuel assemblies with water level lower than 56'- 10" elevation during rerack operation.~~

(*) ~~**~~The requirements of this specification may be suspended for more than 4 hours hours to perform maintenance provided a safety evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

3/4.9.12 HANDLING OF SPENT FUEL CASK

LIMITING CONDITION FOR OPERATION

3.9.12 The handling of spent fuel cask shall be limited to the following conditions:

1) The spent fuel cask shall not be moved into the spent fuel pit until all the spent fuel in the pit has decayed for a minimum of one thousand five hundred twenty-five (1,525) hours.* DELETE

2) Only a single element cask may be moved into the spent fuel pit.

3) A fuel assembly shall not be removed from the spent fuel pit in a shipping cask until it has decayed for a minimum of one hundred twenty (120) days.

APPLICABILITY: During movement of spent fuel cask in the spent fuel storage area.

ACTION:

With the requirement of the above specification not satisfied, suspend all movement of the spent fuel cask within the spent fuel storage area.

SURVEILLANCE REQUIREMENTS

4.9.12.1 The following required decay times of the spent fuel assemblies shall be determined prior to the movement of a spent fuel cask by verification of date and time the spent fuel assemblies were placed into the spent fuel pit:

a. 1525 hours of decay of all spent fuel assemblies in the spent fuel pit for movement of a spent fuel cask into the spent fuel pit.

b. 120 days of decay of the spent fuel assembly in the spent fuel cask prior to removal of the spent fuel cask from the spent fuel pit.

4.9.12.2 Prior to any operations involving spent fuel cask movement into the spent fuel pit, verify only a single element cask will be moved into the spent fuel pit.

9 4.~~8~~.12.3 The spent fuel cask crane interlock shall be demonstrated OPERABLE within 7 days of crane operation and at least once per 7 days (7 days is maximum time between tests; specification 4.0.2 does not apply here) when the crane is being used to maneuver the spent fuel cask.

~~*The spent fuel cask can be moved into the Unit 4 spent fuel pit after a minimum decay of 1000 hours until the new two-region high density spent fuel racks are installed.~~

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. ~~Fuel assemblies containing more than 4.1 weight percent of U-235 shall not be placed in the single region spent fuel storage racks. After installation of the two-region high density spent fuel racks,~~ T ~~t~~he maximum enrichment loading for the fuel assemblies in the spent fuel racks shall ~~be~~ not exceed 4.5 weight percent of U-235.

b. The minimum boron concentration in the Spent Fuel Pit shall be 1950 ppm.

c. ~~*~~ Storage in Region II of the Spent Fuel Pit shall be further restricted by burnup and enrichment limits specified in Table 3.9-1.

DELETE

~~d.* During the re-racking operation only, fuel that does not meet the burnup requirement for normal storage in Region II may be stored in Region II in a checkerboard arrangement (i.e., no fuel stored in adjacent spaces).~~

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With either ~~any of~~ condition~~s~~ a, or c ~~or d~~ not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

b. With boron concentration in the Spent Fuel Pit less than 1950 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 1950 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.14 The boron concentration of the Spent Fuel Pit shall be verified to be 1950 ppm or greater at least once per month.

~~*These requirements are applicable only after installation of the new two-region high density spent fuel racks.~~

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GAS DECAY TANK SYSTEM (as measured in the inservice gas decay tank) 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 inservice gas decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the inservice gas decay tank greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the gas decay tanks and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the inservice gas decay tanks shall be determined to be within the above limits by continuously* monitoring the waste gases in the inservice gas decay tank with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-8 of Specification ~~3.3.3.7.~~

3.3.3.6

3.3.8

*When continuous monitoring capability is inoperable, Table ~~3.3-9~~ allows the use of grab samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

ACTION (Continued) broad leaf

c. With milk or ~~fresh leafy~~ vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

SPENT FUEL STORAGE (Continued)

in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.9-1, placement in a Region II cell is authorized. These positive controls assure the fuel enrichment limits assumed in the safety analyses will not be exceeded.*

DELETE

~~*This Technical Specification is applicable upon installation of the new two-region high density spent fuel racks.~~

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison and shall be maintained with:

~~a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.55% Δk/k for uncertainties for single region spent fuel storage racks~~.

a. ~~b.~~ A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance in region 1 of 0.97% Δk/k and in region 2 of 1.96% Δk/k for uncertainties for two region fuel storage racks.

b. ~~c.~~ ~~A nominal 13.7 inch center-to-center distance between fuel assemblies placed in the single region storage racks~~. A nominal 10.6 inch center-to center distance for Region 1 and 9.0 inch center-to-center distance for Region 2 for two region fuel storage racks.

~~d. Fuel assemblies stored in the single region spent fuel storage racks shall contain no more than 4.1 weight percent of U-235~~.

c. ~~e.~~ ~~After installation of the two region high density spent fuel storage racks,~~ T~~t~~he maximum enrichment loading for fuel assemblies is 4.5 weight percent of U-235.

5.6.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

DELETE

5.6.1.3 Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks*. Administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

DRAINAGE

5.6.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than ~~621** fuel assemblies in one region storage racks or~~ 1404 in two region storage racks

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

~~*During rack installation, it will be necessary to temporarily store Region I fuel in the Region II spent fuel racks. Administrative controls will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks~~.

~~**The fuel assembly storage capacity for Unit 4 single region storage racks is 614~~.

[illegible]

[illegible]

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager - Nuclear shall be responsible for overall unit operation of both units and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Plant Supervisor - Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to ~~an~~ , and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The President-Nuclear Division shall have corporate responsibility for overall plant nuclear safety, and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The Plant Manager-Nuclear shall be responsible for overall plant safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the health physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

RESPONSIBILITIES (Continued)

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board;

f. Review of all REPORTABLE EVENTS;

g. Review of reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment or systems that affect nuclear safety.

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager - Nuclear or the Chairman of the Company Nuclear Review Board;

i. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Chairman of the Company Nuclear Review Board;

j. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL;

k. Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board.

6.5.1.7 The PNSC shall:

a. Recommend in writing to the Plant Manager - Nuclear approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation and items considered under Specification 6.5.1.6i through k.

INSERT

b. Provide written notification within 24 hours to the Plant Manager-Nuclear, President-Nuclear Division and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Manager - Nuclear; however, the Plant Manager - Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

[illegible]

PROCEDURES AND PROGRAMS (Continued)

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed in accordance with Specification 6.5.3 and approved by the Plant Manager-Nuclear or the department head of the responsible department 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 Safety Injection System, Chemical and Volume Control System, and the Containment Spray System. The program shall include the following:

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

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

b. In-Plant Radiation Monitoring

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

(1) Training of personnel,

(2) Procedures for monitoring, and

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

c. Secondary Water Chemistry

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

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

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

PROCEDURES AND PROGRAMS (Continued)

~~c. Secondary Water Chemistry.~~

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

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

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

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

d. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(1) Training of personnel,

(2) Procedures for sampling and analysis, and

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

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.3 Routine Annual Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of the following year.

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

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

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

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

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

RADIATION PROTECTION PROGRAM ~~RECORD RETENTION~~ (Continued)

maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates 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 (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. 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 monitoring device which continuously indicates the radiation dose rate in the area; or

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 levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the Health Physics Shift Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift supervisor on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area.

RADIATION PROTECTION PROGRAM