ATTACHMENT I

REQUESTED CHANGE

FOR

VIRGIL C. SUMMER STATION

TECHNICAL SPECIFICATION 3.2.3

PAGES AFFECTED

The following is an explanation for the requested change to Figure 3.2.3.

PAGE	EXPLANATION
3/4 2-10	Current estimates of total reactor coolant systems (RCS) flow at the Virgil C. Summer Nuclear Station indicate that the measured flow, when measurement uncertainties of 3.5% are included, may be less than the Thermal Design (TD) flow used in the plant accident analyses. The measured flow will be determined at a future date by a flow calorimetric with the plant operating above 50% power. The finite time period allowed for restoring RCS flow to above the minimum required value in the present Technical Specification is based on the assumption that the low flow is due to time limited local effects, e.g. grid voltage/frequency dips. The concern at Summer is not with low flow due to time limited local effects, but rather with the measured RCS flow being less than TD flow when flow measurement uncertainties of 3.5% are included. The Technical Specification is quite specific in the action to be taken if the measured flow is less than TD flow. The results of the required action is that Thermal Power will be limited to less than 5% of Rated Thermal Power. This is because the Technical Specification does not recognize the possibility of a long term reduction in flow, nor the various trade-offs allowed by the relationships between flow, DNB, and core power. These trade-offs can be used to justify continued operation at some reduction in the maximum allowed power if the measured RCS flow is less than required. It is widely recognized that the relationships between core power, flow, and DNB are: $\frac{\partial \text{ Flow}}{\partial \text{ DNB}} = \frac{1\%}{1\%}$ (Eq. 1) $\frac{\partial \text{ Power}}{\partial \text{ DNB}} = \frac{1\%}{1.8\%}$ (Eq. 2) Thus the relationship between Power and Flow is: $\frac{\partial \text{ Power}}{\partial \text{ Flow}} = \frac{1\%}{1.8\%}$ (Eq. 3)

EXPLANATION

Based on a conservative assumption that the measured RCS flow will be no lower than 95% of TD flow, it is requested that a region of acceptable operation be added to Figure 3.2.3 for:

95% TD Flow $\leq$ RCS Flow $\leq$ 100% TD Flow (Eq. 4)

Based on the relationship given by Equation 3, it is recommended that the maximum power level for this region be restricted to less than or equal to 95% of Rated Thermal Power (2636 MW_t). This restriction of core power is the equivalent of an RCS flow increase of approximately 9% in terms of DNB margin. Since in this region the flow deficit is 5% or less, the power restriction will result in a minimum increase of approximately 4% in terms of DNB margin. Operation of the plant in this region within the specified power restriction results in no increase in Tavg, thus there is no temperature impact on the DNB margin.

The Technical Specifications and accident analysis results have been evaluated to determine the impact of operating within the defined new region of Figure 3.2-3 with the imposed restrictions. In all cases sufficient margin exists to allow continued plant operations. No Technical Specification limits require modification, including core limits, OTΔT, OPΔT, and Power Range Neutron Flux High Setpoint.

The core limits remain the same due to the increased margin to DNB afforded by the power reduction and interpretation that they will be valid for a maximum power level of 2636 MW_t instead of the design 2775 MW_t. This implies that under these conditions 2636 MW_t should be considered to be 100% of Rated Thermal Power for Figure 2.1-1. With this restriction applied to the Safety limits there is no change in the core limits, thus the OTΔT and OPΔT trip setpoints remain unchanged. Utilizing the latest Westinghouse data, the uncertainty in the instrumentation for the Power Range Neutron Flux High trip function is 4.7% span (or 5.7% RTP). With a normal assumption of reactor trip at 109% RTP the uncertainty analysis verifies that a trip will take place at 109% RTP plus 5.7% uncertainty or 114.7% RTP. A 5% reduction in RCS flow requires a trip at 115.2% RTP. Therefore, adequate margin exists in the instrumentation such that no change in the nominal setpoint is necessary.

EXPLANTION

Obviously, if the measured RCS flow (including 3.5% uncertainties) is equal to or greater than Thermal Design flow, operation will be in the acceptable region of the present Figure 3.2-3 and the requirements of this specification will remain unchanged. The addition of the new region to Figure 3.2-3 is only requested to preclude a needless reduction to 5% RTP if the measured RCS flow is found to be low.

ACTION: (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R_1, R_2 and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R_1, R_2 and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4 0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R_1, R_2 shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained values of R_1 and R_2, obtained per Specification 4.2.3.2, are assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.3.5 The RCS total flow rate shall be determined by measurement at least once per 18 months.

ATTACHMENT II

REQUESTED CHANGES

FOR

VIRGIL C. SUMMER STATION

TECHNICAL SPECIFICATIONS

3.3.3.7
3.7.9.2
3.7.9.4
3.7.9.5

PAGES AFFECTED

The following is a brief explanation for the requested changes to the Fire Protection Technical Specifications.

PAGE	EXPLANATION
3/4 3-62 thru 3/4 3-66d	The corrections to the Fire Detection Instruments Table 3.3-11 are necessary to reflect changes in the installation.
3/4 7-28	Elevations have been eliminated from the identification of the spray and sprinkler system because they do not accurately describe areas protected. The name is sufficient to identify the system.
3/4 7-32	Additional fire hose stations have been added to Table 3.7-5 to ensure adequate fire protection of safety related equipment.
3/4 7-34	An additional yard fire hydrant and hose house have been added to Table 3.7-6 to ensure adequate fire protection for the diesel generators.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
1. REACTOR BUILDING				
Zone 1 Elev. 412'				
Room 12-01 NE		2		3
Room 12-01 SE		2		3
Room 12-01 SW		2		3
Room 12-01 NW		2		3
Zone 2 Elev. ~~412'~~ 436'				
Room 12-03		1		2
Room 12-07		1		2
Room 12-08		1		2
Zone 3 Elev. 436'				
Room 36-01 NE		2		3
Room 36-01 SE		2		3
Room 36-01 NW		2		3
Room 36-01 SW		2		3
Zone 4 Elev. 462'				
Room 12-03		1		2
Room 12-07		1		2
Room 12-08		1		2
Zone 000 Elev. ~~426'~~ 495'				
Room 12-08		1		2
Zone 000 Elev. ~~463'~~ 514'				
Room 63-01 SE		2		3
Room 63-01 SW		1		2
Zone PPP Elev. ~~412'~~ 495'				
Room 12-03		1		2
Room 12-07		1		2
Zone PPP Elev. ~~463'~~ 514'				
Room 63-01 NE		1#		1 & 2#
Room 63-01 NW		3#		6 & 5#
2. CONTROL BUILDING				
Zone ~~F~~ A Elev. 463'				
~~Room 63-02~~		~~1**~~		~~1**~~
Room 63-03		3 & **		4 & **
Room 63-04		1**		2**

(A) on next Page

SUMMER - UNIT 1

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
2. CONTROL BUILDING (Cont.)				
Zone A Elev. 448'				
Room 48-02 (Cable Spreading Room-Upper)		17**		18**
Zone A Elev. 436'				
~~Room 36-02~~		~~1**~~		~~1**~~
Room 36-03		3**		4**
Room 36-04		1**		2**
Zone B Elev. 425'				
Room 25-02 (Cable Spreading Room-Lower)		10**		11**
Zone B Elev. 412'				
Room 12-04 ← Room 12-03		3**		4**
← Room 12-04A		1**		2**
		1**		1**
Zone B Elev. 400'				
~~Below~~ Room ~~12-05~~ 00-01		1**		1**
← Room 00-01A		1**		1**
(A) → Zone J Elev. 436'				
Room 36-01	2**		3**	
Room 36-02	2**		3**	
Zone J Elev. 448'				
Room 48-01	21**		22**	
Zone K Elev. 412'				
Room 12-02	2**		3**	
~~Room 12-03~~	~~3**~~		~~4**~~	
Zone K Elev. 425'				
Room 25-01	21**		22**	
Room 25-03	1**		2**	
Room 25-04	1**		2**	
Zone L Elev. 436'				
Room 36-11	~~23***~~	4#	~~24***~~	7#
Zone M Elev. 436'				
Room 36-10	~~14***~~	1	~~15***~~	2
Zone N Elev. 448'				
Room 48-02 (Cable Spreading Area For HVAC Cabinets) & MCB		13 ~~11~~#		25 ~~21~~#
Zone P Elev. 463'				
Room 63-05 (Main Control Board)		3#		6#
Zone R Elev. 463'				
Room 63-05 (HVAC Control Board)		1		1
Zone O Elev. 463'				
Room 63-02	1**		2**	

Zone I Elev. 436
Room 36-10 2*** 3***
Room 36-11 2*** 23***
Zone U Elev. 436' 2*** 3***
Room 36-09
Zone I Elev. 436' 5# 10#
Room 36-11

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
2. CONTROL BUILDING (Cont.)				
Zone S Elev. 448'				
Room 48-02 (Cable Spreading Area for HVAC Cabinets) & MCB		12 ~~8~~#		23 ~~17~~#
Zone VV Elev. 436'				
Room 36-09		1		2
Zone WW Elev. 463'				
Room 63-05		4#		8#
Room 63-06		1		1
Room 63-07		1		2
Room 63-09		1		1
Room 63-10		1		1
Room 63-11		1		1
Room 63-12		1		1
Room 63-13		1		1
Zone XX Elev. 463'				
Room 63-14		1		1
Room 63-15		1		1
Room 63-16		1		1
Room 63-17		1		1
Room 63-18		1		2
Room 63-19		1		1
Room 63-20		1		1
Room 63-21		1		1
Room 63-22		1		2
Room 63-23		1		1
Zone NNN Elev. 482'				
Room 82-01		4#		8#
Room 82-02		4#		7#
Room 82-03		2		3
Room 82-04		1		2
Zone TTT Elev. 412'				
Room 12-11		1		2
~~3. NORTH PENETRATION ACCESS AREA~~				
~~Zone EE Elev. 436'~~				
~~Room 36-01~~		~~2~~		~~3~~
~~Zone YY Elev. 412'~~				
~~Room 12-01~~		~~2#~~		~~4#~~

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
INTERMEDIATE BUILDING (Cont.)				
3. EAST PENETRATION ACCESS AREA				
Zone JJ Elev. 412'				
Room 12-02		5#		10#
Zone LL Elev. 436'				
Room 36-02		5#		10#
4. WEST PENETRATION AREA				
Zone Y Elev. 412'				
Room 12-01		4#		8#
Zone DD Elev. 436'				
Room 36-01		4#		8#
Zone II Elev. 463'				
Room 63-01		2 ~~3~~#		3 ~~8~~#
Room 63-03		3# 2		6# 8
5. INTERMEDIATE BUILDING				
Zone AA Elev. 412' ~~Room 12-02~~		15**		16**
Room 12-02 ~~W~~ {Room 12-02 S		4 ~~22~~** {1**		5 ~~23~~** {1**
Room 12-13 A {Room 12-09		1** {1**		1** {1**
Room 12-13 B		1**		1**
Room 12-13 C		1**		1**
Zone BB Elev. ~~412'~~ Room 12-02		10**		11**
Room 12-02 E		6 ~~17~~**		7 ~~18~~**
Room 12-10		1**		1**
Zone CC Elev. 436'				
Room 36-02 ~~W~~SW		3**		4**
Zone FF Elev. 423'-6"				
Room 236-01		3#		5#
Zone GG Elev. 412'				
Room 12-03		1		1
Room 12-04		1		1
Room 12-05		1		2
Room 12-06		1		2
Room 12-07		1		1
Room 12-08		1		1
Room 12-12		1		1
Room 12-14		1		1
Room 12-15		1		1

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
5.6. INTERMEDIATE BUILDING (Cont.)				
Zone KK Elev. 412'				
Room 12-09		1		1
Zone KK Elev. 423'				
Room 23-01		2#		4#
Room 23-02		2#		4#
Zone KK Elev. 436'				
Room 36-01		2		3
Zone MM Elev. 463'				
Room 63-01		4#		8#
Room 63-02		3#		6#
Room 63-03		1		2
Zone PP Elev. 436'				
Room 36-02		8#		16#
Room 36-03 (CREP)		1		2
Room 36-03 A (CREP)		1		2
Room 36-03 B		1		2
Room 36-04		1		1
Zone MMM Elev. 451'				
Room 51-01		3#		6#
Room 51-02		3#		6#
Room 51-03		1		2
Room 51-04		1		2
Zone FFF Elev. 436'				
Room 36-02		8#		16#
Room 36-02 E		1		1
Zone UUU Elev. 426'				
Room 26-01		2#		4#
Room 26-02		2#		4#
6.7. AUXILIARY BUIDLING				
Zone Q Elev. 436'				
Room 36-18		2#		4#
Zone HH Elev. 463'				
Room 63-01		1		2
Room 63-04		1		1
Room 63-06		1		2
Room 63-17		1		1

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
6.8. AUXILIARY BUILDING (Cont.)				
Zone QQ Elev. 452'				
Room 52-01		1		1
Room 52-02		1		1
Zone RR Elev. 463'				
Room 63-16		2**		3**
Room 63-19		1**		1**
Zone ZZ Elev. 374'				
Room 74-12		1		1
Room 74-16		2		3
Room 74-17		2		3
Zone AAA Elev. 388'				
Room 88-23		1		1
Room 88-24		1		1
Room 88-25		1		1
Zone AAA Elev. 397'				
Room 97-02		3#		6#
Room 97-02 N		1		1
Room 97-02 S		1		2
Zone BBB Elev. 388'				
Room 88-05		1		2
Room 88-13 N		2		3
Room 88-13 NE		1		1
Room 88-13 S		1		1
Zone BBB Elev. 397'				
Room 97-01		1		2
Zone BBB Elev. 400'				
Room 00-01 W		1		1
Room 00-02 E		2		3
Zone CCC Elev. 463'				
Room 63-09		2		3
Room 63-14		1		2
Zone EEE Elev. 374'				
Room 74-01		1		2
Room 74-07		1		2
Room 74-08		2		3
Room 74-09		3#		5#
Room 74-18		1		1
Zone GGG Elev. 400'				
Room 00-01		3#		6#

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
6.X. AUXILIARY BUILDING (Cont.)				
Zone HHH Elev. 412'				
Room ~~12-05~~ 293-01 W		1		1
Room 12-06		1#		1#
~~Room 12-11~~		4		7
~~Room 12-11 N~~		1		1
~~Room 12-27~~		1		1
Zone HHH Elev. 426'				
Room ~~26-02 K~~ 26-01		5#X		10#
Room 26-02 ~~K~~ W		1		1
Room 26-02 S		1		1
Zone III Elev. 412'				
Room 12-02		1		1
Room 12-03 A		1		1
Room 12-09		1		1
Room 12-11 N		1		1
Room 12-15		1		1
Room 12-18		1		1
Room 12-28		1		1
Room 12-31		1		1
Zone JJJ Elev. 412'				
Room 12-02		1#		1#
Room 12-11		4		7
Room 12-11 N		1		1
Zone LLL Elev. 436'				
Room 36-33 N		2#		3#
Room 36-18		2		3
Zone RRR Elev. 388'				
Room 88-05		1		1
Room 88-13		1		2
Room 88-13 S		1		1
Room 88-16		1		1
Zone WWW Elev. ~~436'~~ 448'				
Room ~~36-33~~ 48-01		3#		5#
Zone WWW Elev. 446'				
Room 46-01		1		1

Zone YY Elev. 412'
Room 12-01

2#

4#

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
6. ~~X~~. AUXILIARY BUILDING (Cont.)				
Zone YYY Elev. 412'				
Room 12-05		1		1
Room 12-27		1		1
Room 12-30		1		1
Zone ZZZ Elev. ~~412'~~ 445'				
Room ~~12-01~~ 45-01		1		1
Zone ZZZ Elev. 436'				
Room 36-01		1		1
Room 36-03		1		2
Room 36-31		1		1
Room 36-33 S		1		2
7. ~~X~~. FUEL HANDLING BUILDING				
Zone EE Elev. 436'				
Room 36-01 W		2		3
Room 36-01 E		1		1
Zone EE Elev. 443'-6"				
Room 436-01		1		1
Zone TT Elev. 463'				
Room ~~36-01~~ 63-01		5 ~~10#~~ 5#		10 ~~20#~~ 10#
Room 63-01 N				
Zone UU Elev. 463'				
Room ~~36-01~~ 63-01		7 ~~10#~~ 5#		13 ~~19#~~ 9#
Room 63-01 S				
8. ~~X~~. SERVICE WATER PUMPHOUSE				
Zone NN Elev. 425'				
Room 25-04		1		2
Room 25-05		3#		6#
Zone OO Elev. 441'				
Room 41-02		1**		1**
Zone OO Elev. 436'				
Room ~~36-02~~ 36-01		8**		9**
Zone DDD Elev. 441'				
Room 41-01		3#		5#
Room 41-01 A		1		1
Zone VVV Elev. 425'				
Room 25-03		1		1

TABLE 3.3-11 (Cont.)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	MINIMUM INSTRUMENTS OPERABLE*		TOTAL NUMBER OF INSTRUMENTS	
	HEAT	SMOKE	HEAT	SMOKE
~~9.~~ 10. DIESEL GENERATOR BUILDING				
Zone DG Elev. 436'				
Room 36-01	1**		2**	
Room 36-02	1**		2**	
Room 36-03	7 2**		8 2**	
Room 36-04	7 2**		8 2**	
Zone DG Elev. 427'				
Room 27-03	1**		1**	
Room 27-04	1**		1**	
Zone KKK Elev. 400'				
Room 00-01		1		1
Room 00-02		1		1
Zone KKK Elev. 427'				
Room 27-01		1		1
Room 27-02		1		1
Room 27-03		2		3
Room 27-04		2		3

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

**Automatically Actuates Preaction Sprinkler System.

***Automatically Actuates Low Pressure CO_2 System.

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-12 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. Additionally if this condition prevails for more than 30 days, in the next semiannual effluent report, explain why this condition was not corrected in a timely manner.

c. The provisions of Specifications 6.9.1.12.b, 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST operations at the frequencies shown in Table 4.3-8.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.9.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1. The fuel storage tank contains at least 150 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-197[illegible], is within the acceptable limits specified in Table 1 of ASTM D975-197[illegible] when checked for viscosity, water and sediment.

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.9.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

c. At least once per 18 months by verifying that:

1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.9.2 The following spray and/or sprinkler systems shall be OPERABLE:

a. Fuel Handling Building Charcoal Filter Plenums. ~~Aux. Bldg. Elev. 463'~~

b. Control Room Emergency Charcoal Filter Plenums. ~~Control Bldg. Elev. 405'~~

c. Diesel Fire Pump Room Wet Pipe Sprinkler System. ~~Circulating Water Intake Structure Elev. 436'~~.

d. Diesel Generator Building Preaction Sprinkler System. ~~Diesel Generator Building Elev. 436' and 427'.~~

e. ~~Cable Spreading Rooms and Cable Chases~~ Control Building Preaction Sprinkler System. ~~Control Building Elev. 463', 448', 436', 425', 412' and 400'.~~

f. Service Water Pump House Preaction Sprinkler System. ~~Service Water Pump House Elev. 436 and 441.~~

g. Intermediate Building Preaction Sprinkler System. ~~Intermediate Building Elev. 412.~~

h. Auxiliary Building Preaction Sprinkler System. ~~463' Elev.~~

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.9.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye connected to the length of hose at the station, the other outlet of the wye connected to a hose of sufficient length to provide coverage for the area unprotected by the inoperable hose station. This shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, within 24 hours. Restore the fire hose station 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 plans and schedule for restoring the station to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.9.4 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by a visual inspection of the fire hose stations accessible during plant operation, to assure all required equipment is at the station.

b. At least once per 18 months by:

1. Removing the hose for inspection and re-racking, and
2. Inspecting all gaskets and replacing any degraded gaskets in the couplings.
3. Visual inspection of the fire hose stations not accessible during plant operation to assure all required equipment is at the station.

c. At least once per 3 years by:

1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.
2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE ~~RACK~~ REEL/RACK IDENTIFICATION**
Auxiliary Building	374	4138, 4140, 4142
"	388	4144, 6764, 4145
"	412	6766, 6761, 6763
"	436	6783, 6769, 4148, 4147
"	463	6784, 6755, 4149
"	397	4143
Reactor Building	412	6778, 6780, 6776
"	436	6777, 6782
"	463	6781, 6779
Fuel Handling Building	436	6802
"	463	6804, 6807
Control Building	463	6809, 6933
"	482	6815, 6810
Intermediate Building	412	4128, 4121, 4122
"	436	4129, 4124, 4123
"	463	4130
Control Building	412	6812, 6813
	425	4067, 4069
	436	6808, 6814
	448	4068, 4070

*List all Fire Hose Stations required to ensure the OPERABILITY of safety related equipment.

**Identified by isolation valve number

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.9.5 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-6 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours. Restore the hydrant or hose house 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 hydrant or hose house to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.9.5 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months (once during March, April or May and once during September, October or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

3. Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-6

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION	HYDRANT NUMBER*	EQUIPMENT ID.
West of Service Water Pumphouse	3	XFX-1N-FS
North of Diesel Generator Building	2	XFX-1M-FS

*These hydrant numbers are the numbers physically indicated on the hydrant houses.

ATTACHMENT III

REQUESTED CHANGE

FOR

VIRGIL C. SUMMER STATION

TECHNICAL SPECIFICATION 3.4.3

PAGES AFFECTED

The following is a brief explanation for the requested change.

PAGE	EXPLANATION
3/4 4-9	As described on Page 22-84 of the Safety Evaluation Report (NUREG-0717), the two backup pressurizer heater groups are supplied from separate independent emergency busses backed by diesel generators. The power for the backup heaters is always from the emergency power supply. This power supply cannot be transferred to a balance-of-plant (normal) power supply. Therefore, Surveillance Requirement 4.4.3.3 is not applicable for the Virgil C. Summer Nuclear Station.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1288 cubic feet, (92% of indicated span) and at least two groups of pressurizer heaters each having a capacity of at least 125 kw.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

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

~~4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.~~

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION and

b. Operating the valve through one complete cycle of full travel.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with the power removed in order to meet the requirements of 3.4.4.a or 3.4.4.b.

ATTACHMENT IV

REQUESTED CHANGE

FOR

VIRGIL C. SUMMER STATION

TECHNICAL SPECIFICATION 3.4.9.3

PAGES AFFECTED

The following is a brief explanation for the requested change.

PAGE	EXPLANATION
3/4 4-34	This change is requested to avoid notifying the Commission and preparing a Special Report each time a PORV is made inoperable in order to perform required surveillance tests or preventive maintenance. It will have the desirable effect of limiting reporting to failures which render the cold overpressure system inoperable.

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum auxiliary spray water temperature differential of 625°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

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 the maximum setpoint defined by Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.7 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. In the event either PORV becomes inoperable for reasons other than surveillance testing or preventive maintenance notify the Commission within 7 days. In the event both PORVs are inoperable, notify the Commission within 24 hours. In both cases 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 cause of the inoperability, plans for restoring the valves to OPERABLE status and any corrective action necessary to prevent recurrence.

b. The provisions of Specification 3.0.4 are not applicable.

ATTACHMENT V

MISCELLANEOUS CORRECTIONS

FOR

VIRGIL C. SUMMER STATION

TECHNICAL SPECIFICATIONS

PAGES AFFECTED

The following is a brief explanation for each miscellaneous corrections requested.

PAGE	EXPLANATION
3/4 2-11	The rod bow penalty curve is plotted incorrectly. The coordinate points are correct.
3/4 4-5	The number for this Specification should be 3.4.1.4.1.
3/4 8-18	Typographical error. Test setpoints for all three reactor coolant pump breakers are the same, 3960 amps.
3/4 9-9	ALTERATIONS -misspelled.
6-3	Typographical error. The correct title is Emergency Coordinator. Also AMD should be ADM as shown.

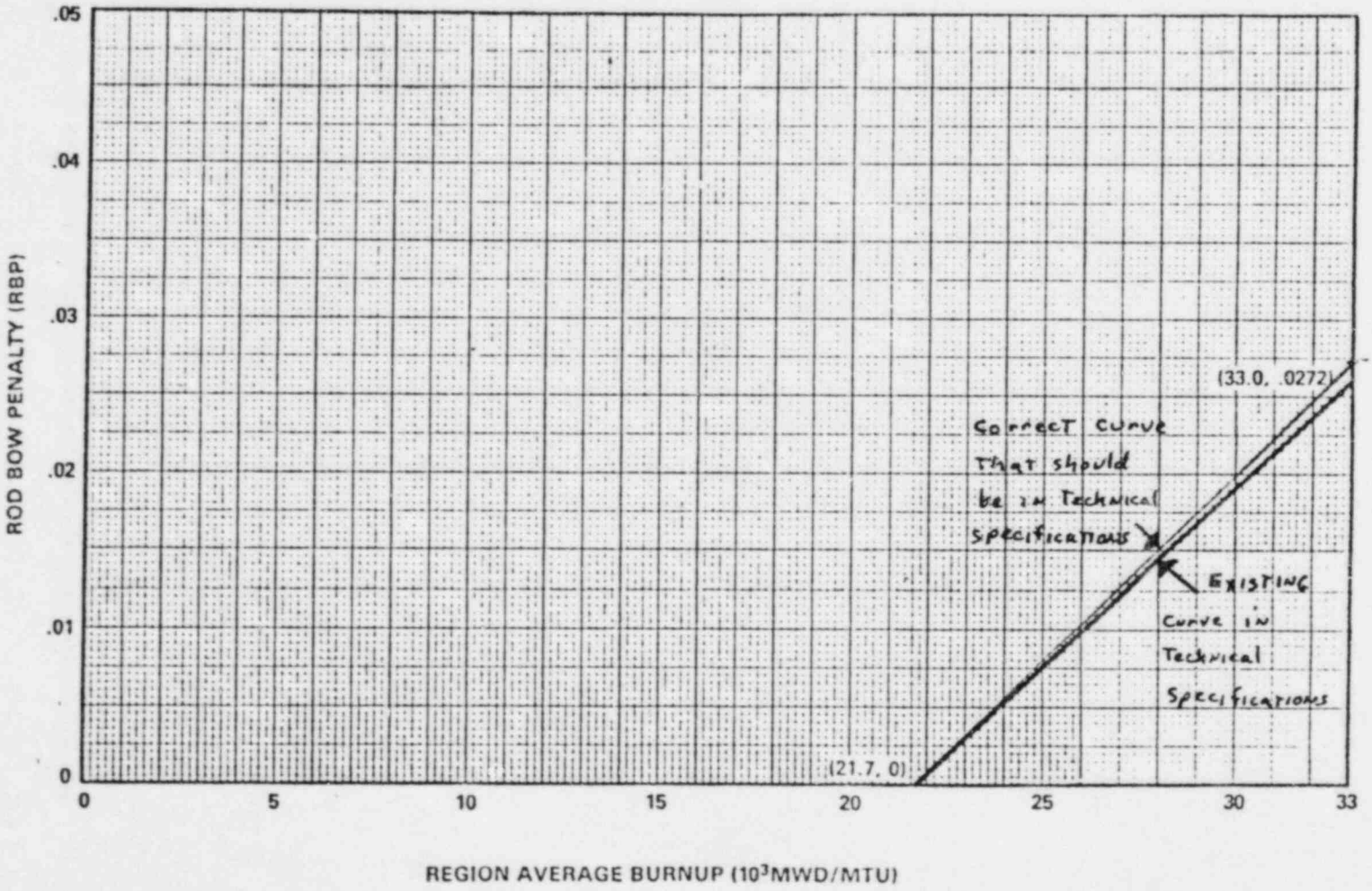

Figure 3.2-4 Rod Bow Penalty as a Function of Burnup

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

4. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10 percent of wide range indication.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

One residual heat removal loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 300°F unless 1) the pressurizer water volume is less than 1288 cubic feet and/or 2) the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

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

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) loops shall be OPERABLE[#] and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

[#] One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

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

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets its manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE TEST SETPOINT CRITERIA

EQUIP NO.-SYS/DESCRIPTION	DEVICE	LOCATION	TEST SETPOINT		RESPONSE TIME
7.2 KV Swgr.					
1) XPP0030A-RC Reactor Coolant Pump A	PRIMARY	XSW1A #9	LONG TIME	3960 Amps	≤ 15.75 Sec.
			INSTANT	5808 Amps	N/A
			GROUND INST.	11 Amps	N/A
BUS1A Normal Feed	BACKUP	XSW1A #5	LONG TIME	5544 Amps	≤ 15.33 Sec.
BUS1A Emergency Feed	BACKUP	XSW1A #3	LONG TIME	5544 Amps	≤ 15.33 Sec.
2) XPP0030B-RC Reactor Coolant Pump B	PRIMARY	XSW1B #7	LONG TIME	~~3690~~ 3960 Amps	≤ 15.75 Sec.
			INSTANT	5808 Amps	N/A
			GROUND INST.	11 Amps	N/A
BUS1B Normal Feed	BACKUP	XSW1B #5	LONG TIME	5544 Amps	≤ 15.33 Sec.
BUS1B Emergency Feed	BACKUP	XSW1B #3	LONG TIME	5544 Amps	≤ 15.33 Sec.
3) XPP0030C-RC Reactor Coolant Pump C	PRIMARY	XSW1C #3	LONG TIME	3960 Amps	≤ 15.75 Sec.
			INSTANT	5808 Amps	N/A
			GROUND INST.	11 Amps	N/A
BUS1C Normal Feed	BACKUP	XSW1C #9	LONG TIME	5544 Amps	≤ 15.33 Sec.
BUS1C Emergency Feed	BACKUP	XSW1C #13	LONG TIME	5544 Amps	≤ 15.33 Sec.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Reactor Building Purge Supply and Exhaust Isolation System inoperable, close each of the Purge and Exhaust penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that Reactor Building Purge Supply and Exhaust isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels, and by verifying that isolation occurs on the 36-inch lines of the Purge Supply and Exhaust Isolation System on a high radiation test signal from the reactor building manipulator crane area channels.

3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel or the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 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 movement of fuel assemblies or control rods.

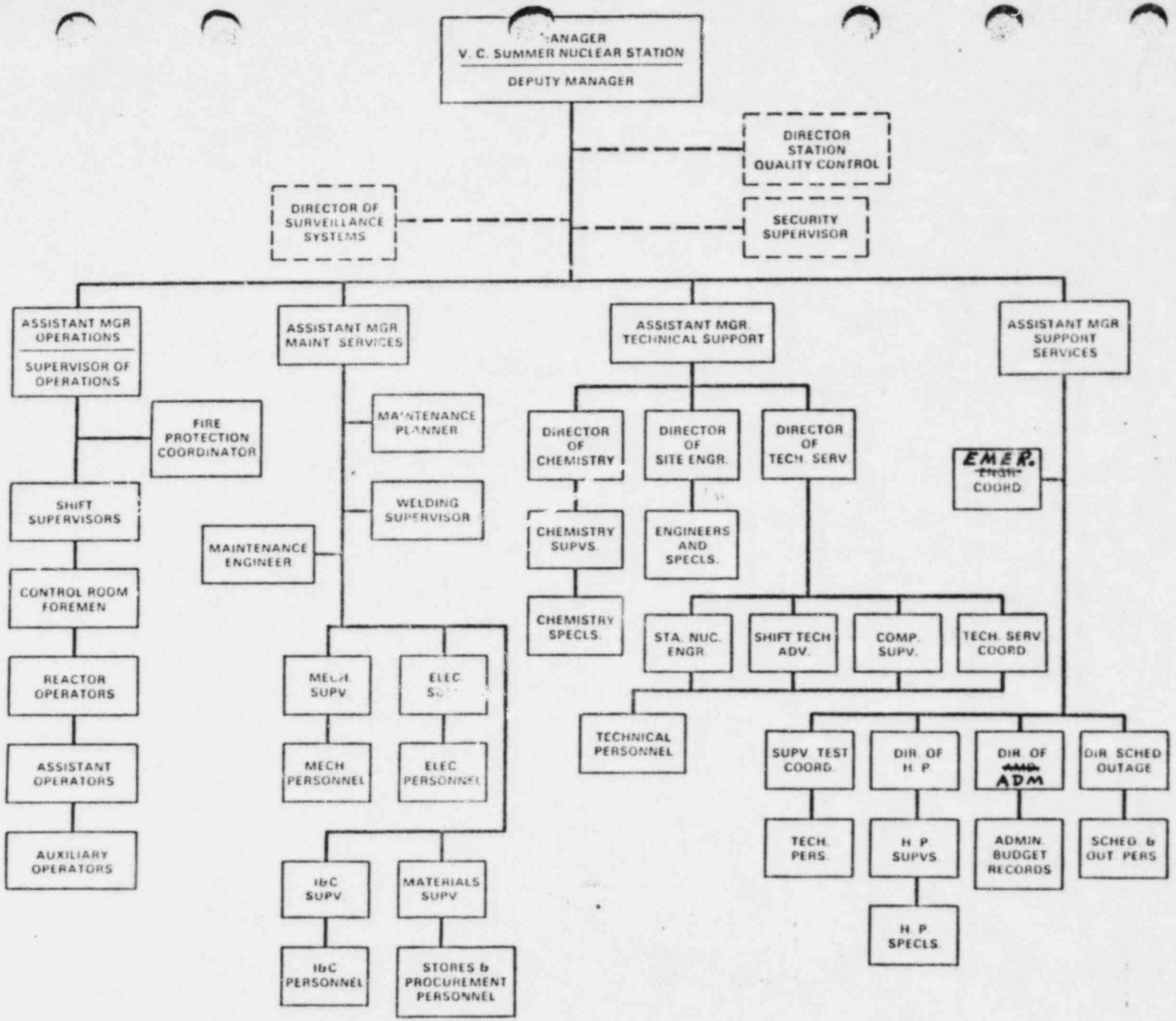

Figure 6.2-2
Virgil C. Summer Nuclear Station
Functional Organization

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SUMMER UNIT 1

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

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1
CRF - Control Room Foreman with a Senior Reactor Operators License on Unit 1
RO - Individual with a Reactor Operators License on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

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

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

ATTACHMENT VI

REQUESTED CHANGE

FOR

VIRGIL C. SUMMER STATION

TECHNICAL SPECIFICATION 3.6.4

PAGES AFFECTED

The following is a brief explanation for the requested changes to Table 3.6-1 of Technical Specification 3.6.4 These changes exclude 14 additional containment isolation valves from Type C testing. Also enclosed is a corresponding change to Table 6.2-53a of the FSAR which will be included in the next amendment. These changes have been discussed with the NRC reviewer, Mr. Hearn.

PAGE	EXPLANATION
3/4 6-20a	It is requested that valves in the reactor coolant pump seal injection lines be excluded from Type C testing because these lines are not isolated during an accident and remain filled and pressurized whenever there is water in the reactor coolant system.
3/4 6-20b	It is requested that the eight motor operated valves located outside containment on the safety injection lines be excluded from Type C testing because in each line there are redundant check valves in series inside containment to prevent leakage and during an accident the lines remain filled and pressurized by the safety injection pumps.
3/4 6-20c	See explanation for page 3/4 6-20a.

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. REACTOR BUILDING PURGE SUPPLY. AND EXHAUST ISOLATION (Continued)		
4. 0002B-AH	Reactor Building Purge Exhaust	5
5. 6056-HR	Alternate Reactor Building Purge Supply Line	5
6. 6057-HR	Alternate Reactor Building Purge Supply Line	5
7. 6066-HR	Alternate Reactor Building Purge Exhaust Line	5
8. 6067-HR	Alternate Reactor Building Purge Exhaust Line	5
D. MANUAL (1)		
1. 8767-DN	Demineralized Water Line	N/A
2. 8768-DN	Demineralized Water Line	N/A
3. 6772-FS	Fire Service Hose Reel Supply	N/A
4. 6773-FS	Fire Service Hose Reel Supply	N/A
5. 2679-IA	Breathing Air Supply Line	N/A
6. 2680-IA	Breathing Air Supply Line	N/A
7. 6587-NG	Nitrogen Supply To Steam Generators	N/A
8. 8090A-RC	Dead Weight Tester	N/A
9. 8090B-RC	Dead Weight Tester	N/A
10. 2912-SA	Reactor Building Service Air	N/A
11. 6671-SF	Refueling Cavity Drain Line	N/A
12. 6672-SF	Refueling Cavity Drain Line	N/A
13. 6697-SF	Refueling Cavity Fill Line	N/A
14. 6698-SF	Refueling Cavity Fill Line	N/A
15. 7135-WL	Reactor Coolant Drain Tank Discharge To Waste	N/A
E. REMOTE MANUAL (2)		
1. 9602-CC	Component Cooling To R. C. Pumps	N/A
2. 8102A-CS #	Seal Injection To Reactor Coolant Pump A	N/A
3. 8102B-CS #	Seal Injection To Reactor Coolant Pump B	N/A
4. 8102C-CS #	Seal Injection To Reactor Coolant Pump C	N/A

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
E. REMOTE MANUAL (Continued)		
5. 8107-CS	Charging Line To Regenerative Heat Exchange	N/A
6. 6050B-HR	Hydrogen Analyzer Return Line	N/A
7. 6051A-HR	Hydrogen Analyzer Supply Line	N/A
8. 6051B-HR	Hydrogen Analyzer Supply Line	N/A
9. 6051C-HR	Hydrogen Analyzer Supply Line	N/A
10. 6052A-HR	Hydrogen Analyzer Return Line	N/A
11. 6052B-HR	Hydrogen Analyzer Return Line	N/A
12. 6053A-HR	Hydrogen Analyzer Supply Line	N/A
13. 6053B-HR	Hydrogen Analyzer Supply Line	N/A
14. 8701A-RH	RHR Pump Suction From Reactor Coolant Loop A	N/A
15. 8701B-RH	RHR Pump Suction From Reactor Coolant Loop C	N/A
16. 8801A-SI #	Boran Injection Tank To Reactor Coolant Loops	N/A
17. 8801B-SI #	Boran Injection Tank To Reactor Coolant Loops	N/A
18. 8811A-SI	RHR Pump A Suction From Recirculation Sump	N/A
19. 8811B-SI	RHR Pump B Suction From Recirculation Sump	N/A
20. 8884-SI #	High Head Safety Injection To Reactor Coolant Loops	N/A
21. 8885-SI #	High Head Safety Injection To Reactor Coolant Loops	N/A
22. 8886-SI #	High Head Safety Injection To Reactor Coolant Loops	N/A
23. 8888A-SI #	Low Head Safety Injection To Reactor Coolant Loops	N/A
24. 8888B-SI #	Low Head Safety Injection To Reactor Coolant Loops	N/A
25. 8889-SI #	Low Head Safety Injection To Reactor Coolant Loops	N/A
26. 3003A-SP	Supply To Reactor Building Spray Nozzles	N/A
27. 3003B-SP	Supply To Reactor Building Spray Nozzles	N/A
28. 3004A-SP	Spray Pump A Suction From Recirculation Sump	N/A
29. 3004B-SP	Spray Pump B Suction From Recirculation Sump	N/A
30. 3103A-SW	Service Water From Reactor Building Cooling Unit A	N/A
31. 3103B-SW	Service Water From Reactor Building Cooling Unit B	N/A
32. 3106A-SW	Service Water To Reactor Building Cooling Unit A	N/A
33. 3106B-SW	Service Water To Reactor Building Cooling Unit B	N/A
34. 3110A-SW	Service Water To Reactor Building Cooling Unit A	N/A
35. 3110B-SW	Service Water To Reactor Building Cooling Unit B	N/A

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER F. CHECK		FUNCTION	MAXIMUM ISOLATION TIME (SEC)
1.	7541-AC	CRDM Coolant Water Inlet Line	N/A
2.	7544-AC	CRDM Coolant Water Outlet Line	N/A
3.	9570-CC	Component Cooling To R. C. Pump Bearings	N/A
4.	9689-CC	Component Cooling From R. C. Pump Bearings	N/A
5.	8103-CS	Reactor Coolant Pump Seal Water Return	N/A
6.	8368A-CS #	Seal Injection To R. C. Pump A	N/A
7.	8368B-CS #	Seal Injection To R. C. Pump B	N/A
8.	8368C-CS #	Seal Injection To R. C. Pump C	N/A
9.	8381-CS	Charging Line To Regenerative Heat Exchanger	N/A
10.	6799-FS	Fire Service Deluge To Charcoal Filters	N/A
11.	2661-IA	Instrument Air Supply To Reactor Building	N/A
12.	6588-NG	Nitrogen Supply To Steam Generators	N/A
13.	8046-RC	Pressurizer Relief Tank Makeup Water Line	N/A
14.	2913-SA	Service Air Supply To Reactor Building	N/A
15.	3009A-SP	Supply To Reactor Building Spray Nozzles	N/A
16.	3009B-SP	Supply To Reactor Building Spray Nozzles	N/A
17.	8947-SI	Accumlator Nitrogen Supply	N/A
18.	8861-SI	Fill Line To Accumulators	N/A

Valve not subject to Type "C" leakage test.

(1) Manual valves may be opened on an intermittent basis under administrative control.

(2) Remote manual valve positions are maintained by administrative control.

TABLE 6.2-53a (Continued)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test: Inside Containment	Type C Leakage Test: Outside Containment	
210 (7)	Reactor Building Leak Rate Test Pressure Sensing Line	LR	No (2)	No (2)	
211 (7)	Reactor Building Leak Rate Test Blowdown	LR	No (2)	No (2)	22
212 (7)	Reactor Building Leak Rate Test Blowdown	LR	No (2)	No (2)	
213	Emergency Feedwater C	EF	N/A	No (3)	3
214	Spare	-	N/A	N/A	
215	Spare	-	N/A	N/A	
216 (7)	Reactor Building Leak Rate Test Pressurization Line	LR	No (2)	No (2)	22
217	Reactor Building Pressure Sensing	-	N/A	N/A	
218	Spare	-	N/A	N/A	
219	Steam Generator Blowdown Loop C	BD	N/A	No (3)	
220 (4)	Steam Generator Blowdown Loop C Sampling Line	SS	N/A	No (3)	
221 (7)	Seal Injection to Reactor Coolant Pump Loop C	CS	~~Yes~~ No	~~Yes~~ No	34 ~~22~~
222 (7)	High Head Safety Injection to Reactor Coolant Loops	SI	No (5)	~~Yes~~ No (6)	
223 (4)	Sampling Line from Reactor Coolant Loop C	SS	Yes	Yes	5

TABLE 6.2-53a (Continued)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

AMENDMENT ~~22~~ 34
~~NOVEMBER, 1980~~

Penetration Number	Service	System	Type C Leakage Test		
			Inside Containment	Outside Containment	
224	Steam Generator Blowdown Loop B	BD	N/A	No(3)	
225	Steam Generator Blowdown Loop B Sampling Line	SS	N/A	No(3)	
226 (7)	Residual Heat Removal Pump Suction from Reactor Coolant Loop C	RH	Yes	N/A	
227 (7)	Low Head Safety Injection to Reactor Coolant Loops	SI	~~No~~ No(5)	~~No~~ No (5)	~~22~~ 34
228	Spare	-	N/A	N/A	
229 (7)	Seal Injection to Reactor Coolant Pump Loop B	CS	~~No~~ No	~~No~~ No	34 ~~22~~
230A	Reactor Vessel Level Sensing Line	RC	N/A	N/A	
230B	Reactor Vessel Level Sensing Line	RC	N/A	N/A	19
230C	Reactor Vessel Level Sensing Line	RC	N/A	N/A	
230D&E	Spare	-	N/A	N/A	
231(4)	Demineralized Water	DN	Yes	Yes	5
232	Spare	-	N/A	N/A	
301A	H_2 Analyzer Supply	HR	Yes	Yes	
B	H_2 Analyzer Discharge	HR	Yes	Yes	
302 (4)	Alternate H. B. Purge Line	HR	Yes	Yes	22

TABLE 6.2-53a (Continued)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Services	System	Type C Leakage Test		
			Inside Containment	Outside Containment	
322(7)	Low Head Safety Injection to Reactor Coolant Loops	SI	No(5)	~~Yes~~ NO (6)	~~22~~ 34
323(4)	Accumulator Sampling Line	SS	N/A	Yes	5
324	Breathing Air	IA	Yes	Yes	23
325(7)	Low Head Safety Injection to Reactor Coolant Loop Hot Legs	SI	No(5)	~~Yes~~ No (6)	~~22~~ 34
326	Steam Generator Blowdown Loop A	BD	N/A	No(3)	
327(4)	Spray Pump A Suction from Reactor Building Recirculation Sump	SP	N/A	Yes	
328(4)	Spray Pump F Suction from Reactor Building Recirculation Sump	SP	N/A	Yes	
329(4)	Safety Injection Pump A Suction from Reactor Building Recirculation Sump	SI	N/A	Yes	
330(7)	Component Cooling Water from Reactor Coolant Pump Bearings	CC	Yes	Yes	22
401(7)	Supply to Reactor Building Spray Nozzles Train A	SP	Yes	Yes	
402(4)	Reactor Building Purge Supply	AH	Yes(6)	Yes	
403(7)	Reactor Building Cooling Unit B Supply	SW	N/A	Yes	
404(4)	Fire Service Hose Reel Supply	FS	Yes	Yes	5

TABLE 6.2-53a (Continued)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test		
			Inside Containment	Outside Containment	
405(4)	Sampling Line from Pressurizer	SS	Yes	Yes	5
406A(4)	Dead Weight Tester	RC	N/A	Yes	22
B	Reactor Building Pressure Sensing Line	-	N/A	N/A	
407A(4) B(4)	Radiation Monitor Supply and Return	SS	Yes	Yes	
408(7)	Seal Injection to Reactor Coolant Pump Loop A	CS	~~Yes~~ NO	~~Yes~~ NO	34
409(4)	Charging Line to Regenerative Heat Exchanger	CS	Yes	Yes	22
410(7)	Reactor Coolant Pump Seal Water Return	CS	Yes	Yes	
411(4)	Steam Generator Blowdown Loop A Sampling	SS	N/A	No(3)	
412(7)	High Head Safety Injection to Reactor Coolant Loops	SI	No(5)	~~Yes~~ NO(6)	34
413	Spare	-	N/A	N/A	
414	Spare	-	N/A	N/A	
415(7)	High Head Safety Injection to Reactor Coolant Loop	SI	No(5)	~~Yes~~ NO(6)	~~22~~ 34
416	Spare	-	N/A	N/A	
417A,B,C	Reactor Vessel Level Sensing	RC	N/A	N/A	
417D	Sample Return to PRT	SS	Yes	Yes	23
417E	Spare	-	N/A	N/A	

TABLE 6.2-53a (Continued)

CONTAINMENT ISOLATION VALVES SUBJECTED TO TYPE C LEAKAGE TESTS

Penetration Number	Service	System	Type C Leakage Test		
			Inside Containment	Outside Containment	
418(4)	Reactor Coolant Drain Tank to Vent Header and H2	WL	Yes	Yes	5
419(4)	Refueling Cavity Drain Line	SF	Yes	Yes	5
420(4)	Pressure Relief Tank	RC	Yes	Yes	5
421(4)	Refueling Cavity Fill Line	SF	Yes	Yes	5
422(4)	Pressure Relief Tank Makeup	RC	Yes	Yes	5
423(4)	Reactor Coolant Drain Tank	WL	Yes	Yes	5
424(4)	Reactor Building Sump Drain	ND	Yes	Yes	5
425(4)	Safety Injection Pump B Suction from Reactor Building Recirculation Sump	SI	N/A	Yes	22
426(7)	Boron Injection to Reactor Coolant Loops	SI	No(5)	~~Yes~~ No(6)	34
427(4)	Fire Service Deluge	FS	Yes	Yes	5
428	Main Steam Loop A	MS	N/A	No(2)	
500	Reactor Building Pressure Sensing	-	N/A	N/A	
703	Reactor Building Pressure Sensing	-	N/A	N/A	