Enclosure B

Affected Technical Specification Pages

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(Continued)

BASES: 2.1 (Cont'd)

metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the ECCS initiation setpoint would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists.

G. Main Steam Line Isolation Valve Closure Scram

The isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram setpoint at 10% of valve closure, there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant operation at any power level shall be permitted in accordance with Table 3.1.1. The system response time from the opening of the sensor contact up to and including the opening of the scram solenoid relay shall not exceed 50 milliseconds.

B. During operation with the ratio of MFLPD to FRP greater than 1.0 either:

a. The APRM System gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1 and 2.1.B or

b. The power distribution shall be changed to reduce the ratio of MFLPD to FRP.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Once a day during reactor power operation the maximum fraction of limiting power density and fraction of rated power shall be determined and the APRM system gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1.a and 2.1.B.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating: Refuel (1)	Startup (12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)
1.	Mode Switch in Shutdown		X	X	X	1	A
2.	Manual Scram		X	X	X	1	A
3.	IRM (7-41(A-F))						
	High Flux	≤120/125	X	X	X(11)	2	A
	INOP		X	X	X(11)	2	A
4.	APRM (APRM A-F)						
	High Flux (flow bias)	≤0.66 (W-ΔW)+54% (4)			X	2	A or B
	High Flux (reduced)	≤15%	X	X		2	A
	INOP				X	2(5)	A or B
	Downscale	≥2/125			X	2	A or B
5.	High Reactor Pressure (PT-2-3-55(A-D)(M))	≤1055 psig	X	X	X	2	A
6.	High Drywell Pressure (PT-5-12(A-D)(M))	≤2.5 psig	X	X	X	2	A
7.	Reactor Low (6) Water Level (LT-2-3-57A/B(M)) (LT-2-3-58A/B(M))	≥127.0 inches	X	X	X	2	A
8.	Scram Discharge Volume High Level (LT-3-231(A-H)(M))	≤21 gallons	X	X	X	2 (per volume)	A

TABLE 3.1.1
(Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings And Allowable Deviations	Modes in Which Functions Must be Operating: Refuel (1)	Startup	Run	Minimum No. Operating Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)
9.	Main steamline high radiation (7) (RD-17-230(A-D)/ RM-17-251(A-D)/ RR-17-252)	3x normal background at rated power(8)	X	X	X	2	A or C
10.	Main steamline isolation valve closure (POS-2-80A/ 86A-A1/B1, POS-2-80B/ 86B-A1/B2, POS-2-80C/ 86C-A2/B1, POS-2-80D/ 86D-A2/B2)	≤10% valve closure			X	4	A or C
11.	Turbine control valve fast closure (PS-(37-40))	(9)(10)			X	2	A or D
12.	Turbine stop valve closure (SOVS-5-(1-4))	≤10% valve(10) closure			X	2	A or D

TABLE 3.1.1 NOTES

1. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a) mode switch in shutdown
   b) manual scram
   c) high flux IRM or high flux SRM in coincidence
   d) scram discharge volume high water level

2. Whenever an instrument system is found to be inoperable, the instrument system output relay shall be tripped immediately. Except for MSIV and Turbine Stop Valve Position, this action shall result in tripping the trip system.

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine load and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 30% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

TABLE 4.1.1
(Cont'd)

SCRAM INSTRUMENTATION AND LOGIC SYSTEMS FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION, LOGIC SYSTEMS AND CONTROL CIRCUITS

Instrument Channel	Group[3]	Functional Test[7]	Minimum Frequency[4]
Scram Test Switch (5A-S2(A-D))	A	Trip Channel and Alarm	Each Refueling Outage
First Stage Turbine Pressure - Permissive (PS-5-14(A-D))	A	Trip Channel and Alarm	Every 6 Months

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group[1]	Calibration Standard[4]	Minimum Frequency[2]
High Flux APRM (APRM A-F)			
Output Signal	B	Heat Balance	Once Every 7 Days
Output Signal (Reduced)	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
LPRM (LPRM ND-2-1-104(80))	B[5]	Using TIP System	Every 1000 Equivalent Full Power Hours
High Reactor Pressure	B	Standard Pressure Source	Once/Operating Cycle
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	B	Standard Pressure Source	Once/Operating Cycle
High Water Level in Scram Discharge Volume	B	Water Level	Once/Operating Cycle
Low Reactor Water Level	B	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	(6)	Refueling Outage
High Main Steam Line Radiation	B	Appropriate Radiation Source[3]	Refueling Outage
First Stage Turbine Pressure Permissive (PS-5-14(A-D))	A	Pressure Source	Every 6 Months and After Refueling
Main Steam Line Isolation Valve Closure	A	(6)	Refueling Outage

BASES:

3.1 Reactor Protection System

The reactor protection system automatically initiates a reactor scram to:

1. preserve the integrity of the fuel barrier;
2. preserve the integrity of the primary system barrier; and
3. minimize the energy which must be absorbed, and prevent criticality following a loss of coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance, testing, or calibration.

The reactor protection system is of the dual channel type. The system is made up of two independent logic channels, each having three subsystems of tripping devices. One of the three subsystems has inputs from the manual scram push buttons and the reactor mode switch. Each of the two remaining subsystems has an input from at least one independent sensor monitoring each of the critical parameters. The outputs of these subsystems are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subsystems will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both logic channels is required to produce a reactor scram.

The required conditions when the minimum instrument logic conditions are not met are chosen so as to bring station operation promptly to such a condition that the particular protection instrument is not required; or the station is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions.

When the minimum requirements for the number of operable or operating trip system and instrumentation channels are satisfied, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

Three APRM instrument channels are provided for each protection trip system to provide for high neutron flux protection. APRM's A and E operate contacts in a trip subsystem, and APRM's C an E operate contacts in the other trip subsystem. APRM's B, D, and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration without changing the minimum number of channels required for inputs to each trip system. Additional IRM channels have also been provided to allow bypassing of one such channel. IRM assignment to the bypass switches is described on FSAR Figure 7.5-9 and in FSAR Section 7.5.5.4.

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specification 2.1.

Instrumentation is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

BASES:

4.1 REACTOR PROTECTION SYSTEM

A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B and C. Sensors that make up Group A are the on-off type and will be tested and calibrated at the indicated intervals. Initially the tests are more frequent than Yankee experience indicates necessary. However, by testing more frequently, the confidence level with this instrumentation will increase and testing will provide data to justify extending the test intervals as experience is accrued.

Group B devices utilize an analog sensor followed by an amplifier and bistable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bistable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during start-up and inactive during full-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The ratio of MFLPD to FRP shall be checked once per day to determine if the APRM gains require adjustment. Because few control rod movements or power changes occur, checking these parameters daily is adequate.

3.2 LIMITING CONDITIONS FOR OPERATION

3.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the operational status of the plant instrumentation systems which initiate and control a protective function.

Objective:

To assure the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

When the system(s) it initiates or controls is required in accordance with Specification 3.5, the instrumentation which initiates the emergency core cooling system(s) shall be operable in accordance with Table 3.2.1.

B. Primary Containment Isolation

When primary containment integrity is required, in accordance with Specification 3.7, the instrumentation that initiates primary containment isolation shall be operable in accordance with Table 3.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

The instrumentation that initiates the isolation of the reactor building ventilation system and the actuation of the standby gas treatment system shall be operable in accordance with Table 3.2.3.

4.2 SURVEILLANCE REQUIREMENTS

4.2 PROTECTIVE INSTRUMENT SYSTEMS

Applicability:

Applies to the surveillance requirements of the instrumentation systems which initiate and control a protective function.

Objective:

To verify the operability of protective instrumentation systems.

Specification:

A. Emergency Core Cooling System

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.1.

B. Primary Containment Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.2.

C. Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.3.

3.2 LIMITING CONDITIONS FOR OPERATION

D. Off-Gas System Isolation

During reactor power operation, the instrumentation that initiates isolation of the off-gas system shall be operable in accordance with Table 3.2.4.

E. Control Rod Block Actuation

During reactor power operation the instrumentation that initiates control rod block shall be operable in accordance with Table 3.2.5.

F. Mechanical Vacuum Pump Isolation

1. Whenever the main steam line isolation valves are open, the mechanical vacuum pump shall be capable of being automatically isolated and secured by a signal of high radiation in the main steam line tunnel or shall be manually isolated and secured.

2. If Specification 3.2.F.1 is not met following a routine surveillance check, the reactor shall be in the cold shutdown within 24 hours.

G. Post-Accident Instrumentation

During reactor power operation, the instrumentation that displays information in the Control Room necessary for the operator to initiate and control the systems used during and following a postulated accident or abnormal operating condition shall be operable in accordance with Table 3.2.6.

4.2 SURVEILLANCE REQUIREMENTS

D. Off-Gas System Isolation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.4.

E. Control Rod Block Actuation

Instrumentation and logic systems shall be functionally tested and calibrated as indicated in Table 4.2.5.

F. Mechanical Vacuum Pump Isolation

During each operating cycle, automatic isolation and securing of the mechanical vacuum pump shall be verified while the reactor is shutdown.

G. Post-Accident Instrumentation

The post-accident instrumentation shall be functionally tested and/or calibrated in accordance with Table 4.2.6.

3.2 LIMITING CONDITIONS FOR OPERATION

H. Drywell to Torus ΔP Instrumentation

1. During reactor power operation, the Drywell to Torus ΔP Instrumentation (recorder #1-156-3 and instrument DPI-1-158-6) shall be operable except as specified in 3.2.H.2.

2. From and after the date that one of the Drywell to Torus ΔP instruments is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless the instrument is sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following eighteen hours.

I. Recirculation Pump Trip Instrumentation

During reactor power operation, the Recirculation Pump Trip Instrumentation shall be operable in accordance with Table 3.2.1.

J. Deleted

K. Degraded Grid Protective System

During reactor power operation, the emergency bus undervoltage instrumentation shall be operable in accordance with Table 3.2.7.

4.2 SURVEILLANCE REQUIREMENTS

H. Drywell to Torus ΔP Instrumentation

The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

I. Recirculation Pump Trip Instrumentation

The Recirculation Pump Trip Instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.1.

J. Deleted

K. Degraded Grid Protective System

The emergency bus undervoltage instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.7.

3.2 LIMITING CONDITIONS FOR OPERATION

L. Reactor Core Isolation Cooling System Actuation

When the Reactor Core Isolation Cooling System is required in accordance with Specification 3.5.G, the instrumentation which initiates actuation of this system shall be operable in accordance with Table 3.2.8.

4.2 SURVEILLANCE REQUIREMENTS

L. Reactor Core Isolation Cooling System Actuation

Instrumentation and Logic Systems shall be functionally tested and calibrated as indicated in Table 4.2.8.

TABLE 3.2.1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Core Spray - A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
2	High Drywell Pressure (PT-10-101(A-D)(M))	$\leq$2.5 psig	Note 2
2	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(M))	$\geq$82.5" above top of enriched fuel	Note 2
1	Low Reactor Pressure (PT-2-3-56C/D(M))	300 $\leq$ P $\leq$ 350 psig	Note 2
2	Low Reactor Pressure (PT-2-3-56A/B(M) & PT-2-3-52C/D(M))	300 $\leq$ P $\leq$ 350 psig	Note 2
1	Time Delay (14A-K16A & B)	$\leq$10 seconds	Note 2
2	Pump (P-46-1A/B) Discharge Pressure (PS-14-44(A-D))	$\geq$100 psig	Note 5
1	Auxiliary Power Monitor (LNPX C/D)	--	Note 5
1	Pump Bus Power Monitor (27/3A/B, 27/4A/B)	--	Note 5
1	Trip System Logic	--	Note 5

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
1	Low Reactor Pressure (PT-2-3-56C/D(M))	300 ≤ p ≤ 350 psig	Note 2
2	High Drywell Pressure (PT-10-101(A-D)(M))	≤2.5 psig	Note 2
2	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(S1))	≥82.5" above top of enriched fuel	Note 2
1	Time Delay (10A-K51A & B)	0 seconds	Note 5
1	Reactor Vessel Shroud Level (LT-2-3-73A/B(M))	≥2/3 core height	Note 5
1	Time Delay (10A-K72A & B)	≤60 seconds	Note 5
1	Time Delay (10A-K50A & B)	≤5 seconds	Note 5
1	Low Reactor Pressure (PS-2-128A & B)	100 ≤ p ≤ 150 psig	Note 2
2 per pump	RHR Pump Discharge Pressure (PS-10-105(A-H))	≥100 psig	Note 5
2	High Drywell Pressure (PT-10-101(A-D)(S1))	≤2.5 psig	Note 2

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
1	Time Delay (10A-K45A & B)	$\leq$6 minutes	Note 5
2	Low Reactor Pressure (PT-2-3-56A/B(S1) & PT-2-3-52C/D(M))	300 $\leq$ p $\leq$ 350 psig	Note 2
1	Auxiliary Power Monitor (LNPX C/D)	--	Note 5
1	Pump Bus Power Monitor (27/3A/B, 27/4A/B)	--	Note 5
1	Trip System Logic	--	Note 5

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

High Pressure Coolant Injection System

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
2 (Note 3)	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(S1))	Same as LPCI	Note 5
2 (Note 4)	Low Condensate Storage Tank Water Level (LSL-107-5A/B)	$\geq$ 3%	Note 5
2 (Note 3)	High Drywell Pressure (PT-10-101(A-D)(M))	Same as LPCI	Note 5
1 (Note 3)	Bus Power Monitor (23A-K41)	--	Note 5
1 (Note 4)	Trip System Logic	--	Note 5
2 (Note 7)	High Reactor Vessel Water Level (LT-2-3-72A/B)(S4)	$\leq$177 inches above top of enriched fuel	Note 5

TABLE 3.2.1
(Cont'd)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Automatic Depressurization

Minimum Number of Operable Instrument Channels per Trip System (Note 4)	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
2	Low-Low Reactor Vessel Water Level (LT-2-3-72(A-D)(M))	Same as Core Spray	Note 6
2	High Drywell Pressure (PT-10-101(A-D)(S1))	$\leq$2.5 psig	Note 6
1	Time Delay (2E-K5A/B)	$\leq$120 seconds	Note 6
1	Bus Power Monitor (2E-K1A/B)	--	Note 6
1	Trip System Logic	--	Note 6
2	Time Delay (2E-K16A/B, 2E-K17A/B)	$\leq$8 minutes	Note 6

TABLE 3.2.1
(Cont'd)

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

Recirculation Pump Trip - A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
2	Low-Low Reactor Vessel Water Level (LM-2-3-68(A-D))	$\geq$ 6' 10.5" above top of enriched fuel	Note 2
2	High Reactor Pressure (PM-2-3-54(A-D))	$\leq$ 1150 psig	Note 2
2	Time Delays (2-3-68(A-D)(X))	$\leq$ 10 seconds	Note 2
1	Trip Systems Logic	--	Note 2

TABLE 3.2.1 NOTES

1. Each of the two Core Spray, LPCI and RPT, subsystems are initiated and controlled by a trip system. The subsystem "B" is identical to the subsystem "A".

2. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits. If the channel cannot be tripped by the means stated above, that channel shall be made operable within 24 hours or an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

3. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.

4. One trip system with initiating instrumentation arranged in a one-out-of-two logic.

5. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.

6. Any one of the two trip systems will initiate ADS. If the minimum number of operable channels in one trip system is not available, the requirements of Specification 3.5.F.2 and 3.5.F.3 shall apply. If the minimum number of operable channels is not available in both trip systems, Specifications 3.5.F.3 shall apply.

7. One trip system arranged in a two-out-of-two logic.

TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied (Note 2)
2	Low-Low Reactor Vessel Water Level (LT-2-3-57A/B(S2), LT-2-3-58A/B(S2))	≥82.5" above the top of enriched fuel	A
2 of 4 in each of 2 channels	High Main Steam Line Area Temperature (TS-2-(121-124)(A-D))	≤212°F	B
2/steam line	High Main Steam Line Flow (DPT-2-(116-119)(A-D)(M))	≤140% of rated flow	B
2/(Note 1)	Low Main Steam Line Pressure (PS-2-134(A-D))	≥800 psig	B
2/(Note 6)	High Main Steam Line Flow (DPT-2-116A, 117B, 118C, 119D (S1))	≤40% of rated flow	B
2	Low Reactor Vessel Water Level (LT-2-3-57A/B(M), LT-2-3-58A/B(M))	Same as Reactor Protection System	A
2	High Main Steam Line Radiation (RD-17-230(A-D)/ RM-17-251(A-D)/RR-17-252) (7) (8)	≤3 x background at rated power (9)	B
2	High Drywell Pressure (PT-5-12(A-D)(M))	Same as Reactor Protection System	A
2/(Note 10)	Condenser Low Vacuum (PS-2-11(A-D))	≤12" Hg absolute	A
1	Trip System Logic	--	A

TABLE 3.2.2
(Cont'd)

HIGH PRESSURE COOLANT INJECTION SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
2 per set of 4	High Steam Line Space Temperature (TS-23-(101-104)(B-D))	≤212°F	Note 3
1	High Steam Line d/p (Steam Line Break) (DPIS-23-77/78)	≤195 inches of water	Note 3
4 (Note 5)	Low HPCI Steam Supply Pressure (PS-23-68(A-D))	≥70 psig	Note 3
2	Main Steam Line Tunnel Temperature (TS-23-(101-104)A)	≤212°F	Note 3
1	Time Delay (23A-K48) (23A-K49)	≤35 minutes	Note 3
1	Bus Power Monitor (23A-K38)	--	--
1	Trip System Logic	--	--

TABLE 3.2.2
(Cont'd)

REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied (Note 2)
2	Main Steam Line Tunnel Temperature (TS-13-(79-82)A)	$\leq$212°F	Note 3
1	Time Delay (13A-K41) (13A-K42)	$\leq$35 minutes	Note 3
2 per set of 4	High Steam Line Space Temperature (TS-13-(79-82)(B,C,D))	$\leq$212°F	Note 3
1	High Steam Line d/p (Steam Line Break) (DPIS-13-83/84)	$\leq$195 inches of water	Note 3
4 (Note 5)	Low Steam Supply Pressure (PS-13-87(A-D))	$\geq$50 psig	Note 3
1	Bus Power Monitor (13A-K36)	--	Note 3
1	Trip System Logic	--	Note 3
1	Time Delay (13A-K7) (13A-K31)	3$\leq$ t $\leq$7 seconds	Note 3

TABLE 3.2.3

REACTOR BUILDING VENTILATION ISOLATION & STANDBY GAS TREATMENT SYSTEM INITIATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
2	Low Reactor Vessel Water Level (LT-2-3-57A/B(M), LT-2-3-58A/B(M))	Same as PCIS	Note 1
2	High Drywell Pressure (PT-5-12(A-D)(M))	Same as PCIS	Note 1
1	Reactor Building Vent (RD-17-430A/B, RM-17-452A/B)	$\leq$14 mr/hr	Note 1
1	Refueling Floor Zone Radiation (RD-17-431A/B, RM-17-453A/B)	$\leq$100 mr/hr	Note 1
1	Reactor Building Vent Trip System Logic	--	Note 1
1	Standby Gas Treatment Trip System Logic	--	Note 1
1	Logic Bus Power Monitor (16A-K52/53)	--	Note 1

Note 1 - If the minimum number of operable instrument channels is not available in either trip system for more than 24 hours, the reactor building ventilation system shall be isolated and the standby gas treatment system operated until the instrumentation is repaired.

TABLE 3.2.4

OFF-GAS SYSTEM ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Conditions For Operation Are Not Satisfied
1	Time Delay (Stack Off-Gas Valve Isolation) (15TD & 16TD)	$\leq$ 2 minutes $\leq$ 30 minutes	Note 1
1	Trip System Logic	--	Note 1

Note 1 - At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System		Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
(Note 1)		Startup Range Monitor				
	2	a. Upscale (Note 2)	X	X		$\leq 5 \times 10^5$ cps (Note 3)
	2	(7-40(A-D))	X	X		
		b. Detector Not Fully Inserted (7-11(A-D)(LS-4))				
		Intermediate Range Monitor				
	2	a. Upscale (7-41(A-F))	X	X		$\leq$108/125 Full Scale
	2	b. Downscale (Note 4) (7-41(A-F))	X	X		$\geq$5/125 Full Scale
	2	c. Detector Not Fully Inserted (7-11(E-K)(LS-4))	X	X		
		Average Power Range Monitor (APRM A-F)				
	2	a. Upscale (Flow Bias)			X	$\leq$0.66(W-ΔW)+42% (Note 5)
	2	b. Downscale			X	$\geq$2/125 Full Scale
(Note 9)		Rod Block Monitor (RBM A/B) (Note 6)				
	1	a. Upscale (Flow Bias) (Note 7)			X	$\leq$0.66(W-ΔW)+N (Note 5)
	1	b. Downscale (Note 7)			X	$\geq$2/125 Full Scale
(Note 8)	1 (per volume)	Scram Discharge Volume (LT-3-231A/G)(S1)	X	X	X	$\leq$12 Gallons
	1	Trip System Logic	X	X	X	

TABLE 3.2.6

POST-ACCIDENT INSTRUMENTATION

Minimum Number of Operable Instrument Channels	Parameter	Type of Indication	Instrument Range
2	Drywell Atmospheric Temperature (Note 1) (TE-16-19-30A/B)	Recorder #TR-16-19-45 Meter #TI-16-19-30B	0-350°F 0-350°F
2	Containment Pressure (Note 1) (PT-16-29A/B)	Meter #PI-16-19-12A Meter #PI-16-19-12B	(-15) -(+260) psig (-15) -(+260) psig
2	Torus Pressure (Note 1) (PT-16-19-36A/B)	Meter #PI-16-19-36A Meter #PI-16-19-36B	(-15) -(+65) psig (-15) -(+65) psig
2	Torus Water Level (Note 3) (LT-16-19-10A/B)	Meter #LI-16-19-12A Meter #LI-16-19-12B	0-25 ft. 0-25 ft.
2	Torus Water Temperature (Note 1) (TE-16-19-33A/C)	Meter #TI-16-19-33A Meter #TI-16-19-33C	0-250°F 0-250°F
2	Reactor Pressure (Note 1) (PT-2-3-56A/B)	Meter #PI-2-3-56A Meter #PI-2-3-56B	0-1500 psig 0-1500 psig
2	Reactor Vessel Water Level (Note 1) (LT-2-3-73A/B)	Meter #LI-2-3-91A Meter #LI-2-3-91B	(-200)-0-(+200)" H_2O (-200)-0-(+200)" H_2O
2	Torus Air Temperature (Note 1) (TE-16-19-39/41)	Recorder #TR-16-19-45 (TE-16-19-34) Meter #TI-16-19-41	0-350°F 50-300°F
2/valve	Safety/Relief Valve Position From Pressure Switches (Note 4) (PS-2-71-(1-3)(A-D))	Lights RV-2-71 (A-D)	Closed - Open

TABLE 3.2.6
(Cont'd)

POST-ACCIDENT INSTRUMENTATION

Minimum Number of Operable Instrument Channels	Parameter	Type of Indication	Instrument Range
1/valve	Safety Valve Position From Acoustic Monitor (Note 5) (ZE-2-1A/B)	Meter ZI-2-1A/B	Closed - Open
2	Containment Hydrogen/Oxygen Monitor (Note 1) (SAH-VG-5A/B)	Recorder SR-VG-6A (SI) Recorder SR-VG-6B (SII)	0-30% hydrogen 0-25% oxygen
2	Containment High-Range Radiation Monitor (Note 6) (RD-16-19-1A/B)	Meter RM-16-19-1A/B	1 R/hr-10^7 R/hr
1	Stack Noble Gas Effluent (Note 7) (RD-17-155)	Meter RM-17-155	0.1 - 10^7 mR/hr

TABLE 3.2.7

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

Minimum Number of Operable Instruments	Parameter	Trip Setting	Required Action
2 per bus	Degraded Bus Voltage - Voltage (27/3Z, 27/3W, 27/4Z, 27/4W)	3,700 volts ± 40 volts	Note 1
2 per bus	Degraded Bus Voltage - Time Delay (62/3W, 62/3Z, 62/4W, 62/4Z)	10 seconds ± 1 second	Note 2

TABLE 3.2.7 NOTES

1. If the minimum number of operable instrument channels are not available, the inoperable channel shall be tripped using test jacks or other permanently installed circuits within one hour.

2. If the minimum number of operable instrument channels are not available, reactor power operation is permissible for only 7 successive days unless the system is sooner made operable.

TABLE 3.2.8

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation are not Satisfied
2 (Note 1)	Low-Low Reactor Vessel Water Level (LT-2-3-72A-D)(M))	$\geq$82.5" Above Top of Enriched Fuel	Note 4
2 (Note 2)	Low Condensate Storage Tank Water Level (LT-107-12A/B)(M))	$\geq$3%	Note 4
2 (Note 3)	High Reactor Vessel Water Level (LT-2-3-72A/B)(S3))	$\leq$177" Above Top of Enriched Fuel	Note 4
1	Bus Power Monitor (13A-K36)	--	Note 4
1	Trip System Logic	--	Note 4

TABLE 3.2.8 NOTES

1. One trip system with initiating instrumentation arranged in a one-out-of-two taken twice logic.

2. One trip system with initiating instrumentation arranged in a one-out-of-two logic.

3. One trip system arranged in a two-out-of-two logic.

4. If the minimum number of operable channels are not available, the system is considered inoperable and the requirements of Specification 3.5 apply.

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Core Spray System

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
High Drywell Pressure	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Low Reactor Pressure	(Notes 1 and 4)	Once/Operating Cycle	--
Low Reactor Pressure	(Notes 1 and 4)	Once/Operating Cycle	--
Pump (P-46-1A/B) Discharge Pressure	(Note 1)	Every Three Months	--
Auxiliary Power Monitor	(Note 1)	Every Refueling	Once Each Day
Pump Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Low Pressure Coolant Injection System

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low Reactor Pressure	(Notes 1 and 4)	Once/Operating Cycle	--
High Drywell Pressure	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Reactor Vessel Shroud Level	(Notes 1 and 4)	Once/Operating Cycle	--
Low Reactor Pressure	(Note 1)	Every Three Months	--
RHR Pump Discharge Pressure	(Note 1)	Every Three Months	--
High Drywell Pressure	(Notes 1 and 4)	Once/Operating Cycle	--
Low Reactor Pressure	(Notes 1 and 4)	Once/Operating Cycle	--
Auxiliary Power Monitor	(Note 1)	Every Refueling Outage	Once Each Day
Pump Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

High Pressure Coolant Injection System

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/operating cycle	Once each day
Low Condensate Storage Tank Water Level	(Notes 1 and 4)	Every three months	--
High Drywell Pressure	(Notes 1 and 4)	Once/operating cycle	Once each day
Bus Power Monitor	(Note 1)	None	Once each day
Trip System Logic	Once/operating cycle	Once/Operating cycle (Note 3)	--
High Reactor Vessel Water Level	(Notes 1 and 4)	Once/operating cycle	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Automatic Depressurization System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
High Drywell Pressure	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic (Except Solenoids of Valves)	Once/Operating Cycle (Notes 2 and 11)	Once/Operating Cycle (Note 3)	--

TABLE 4.2.1
(Cont'd)

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Recirculation Pump Trip Actuation System

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
High Reactor Pressure	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle	--

TABLE 4.2.2

MINIMUM TEST AND CALIBRATION FREQUENCIES

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
High Steam Line Area Temperature	(Note 1)	Each Refueling Outage	--
High Steam Line Flow	(Note 1)	Once/Operating Cycle	Once Each Day
Low Main Steam Line Pressure	(Note 1)	Every Three Months	--
Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	--
High Main Steam Line Radiation	(Notes 1 and 7)	Each Refueling Outage	Once Each Day
High Drywell Pressure	(Notes 1 and 4)	Once/Operating Cycle	Once Each Day
Condenser Low Vacuum	(Note 1)	Every Three Months	--
Trip System Logic	Once/Operating Cycle (Note 2)	Once/Operating Cycle (Note 3)	--

TABLE 4.2.3

MINIMUM TEST AND CALIBRATION FREQUENCIES

REACTOR BUILDING VENTILATION AND STANDBY GAS TREATMENT SYSTEM ISOLATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low Reactor Vessel Water Level	(Notes 1 and 4)	Once/Operating Cycle	--
High Drywell Pressure	(Notes 1 and 4)	Once/Operating Cycle	--
Reactor Building Vent Exhaust Radiation	Monthly	Every Three Months	Once Each Day
Refueling Floor Zone Radiation	Monthly	Every Three Months	Once Each Day During Refueling
Reactor Building Vent Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--
Standby Gas Treatment Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--
Logic Bus Power Monitor	(Note 1)	None	Once Each Day

TABLE 4.2.5

MINIMUM TEST AND CALIBRATION FREQUENCIES

CONTROL ROD BLOCK INSTRUMENTATION

Trip Function	Functional Test	Calibration
Startup Range Monitor		
a. Upscale	Notes 4 and 6	Note 6
b. Detector Not Fully Inserted	Note 6	NA
Intermediate Range Monitor		
a. Upscale	Notes 4 and 6	Note 6
b. Downscale	Notes 4 and 6	Note 6
c. Detector Not Fully Inserted	Note 6	NA
Average Power Range Monitor (APRM A-F)		
a. Upscale (Flow Bias)	Notes 1 and 4	Every Three Months
b. Downscale	Notes 1 and 4	Every Three Months
Rod Block Monitor (RBM A/B)		
a. Upscale (Flow Bias)	Notes 1 and 4	Every Three Months
b. Downscale	Notes 1 and 4	Every Three Months
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)
High Water Level in Scram Discharge Volume	Every Three Months	Refueling Outage

TABLE 4.2.7

EMERGENCY BUS UNDERVOLTAGE INSTRUMENTATION

Trip System	Functional Test	Calibration (8)
Degraded Bus Voltage	See Note 10	Once/Operating Cycle

TABLE 4.2.8

MINIMUM TEST AND CALIBRATION FREQUENCIES

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once each day
Low Condensate Storage Tank Water Level	(Note 1)	Once/Operating Cycle	--
High Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	--
Bus Power Monitor	(Note 1)	None	Once each day
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	--

TABLE 4.2 NOTES

1. Initially once per month; thereafter, a longer interval as determined by test results on this type of instrumentation.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart.

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.

4. This instrumentation shall be functionally tested in accordance with Definition G.1.

5. Deleted.

6. Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibration shall be performed prior to or during each startup or controlled shutdown with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when instruments are required to be operable.

7. This instrumentation is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every three months.

8. Functional tests and calibrations are not required when systems are not required to be operable.

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

3.3 LIMITING CONDITIONS FOR OPERATION

3.3 CONTROL ROD SYSTEM

Applicability:

Applies to the operational status of the control rod system.

Objective:

To assure the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

The core loading shall be limited to that which can be made subcritical in the most reactive condition during the operation cycle with the highest worth, operable control rod in its fully withdrawn position and all other operable rods inserted.

2. Reactivity Margin - Inoperable Control Rods

Control rod drives which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to a shutdown condition within 48 hours unless investigation demonstrates that the cause of the failure is not due to a failed control rod drive mechanism collet housing. The control rod directional control valves for inoperable control rods shall be

4.3 SURVEILLANCE REQUIREMENTS

4.3 CONTROL ROD SYSTEM

Applicability:

Applies to the surveillance requirements of the control rod system.

Objective:

To verify the ability of the control rod system to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

Control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate a shutdown margin of 0.25 per cent Δk at any time in the subsequent fuel cycle with the highest worth operable control rod fully withdrawn and all other operable rods inserted.

2. Reactivity Margin - Inoperable Control Rods

Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week. This test shall be performed at least once per 24 hours in the event power operation is continuing with two or more inoperable control rods or in the event power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive mechanism damage has not been ruled out. The surveillance need not be completed within 24 hours if the number

3.3 LIMITING CONDITIONS FOR OPERATION

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δk. If this limit is exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken if such actions are appropriate.

F. If Specification 3.3A through E above are not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.3 SURVEILLANCE REQUIREMENTS

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

BASES: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the Vermont Yankee Core Performance Analysis Report.

5. The Source Range Monitor (SRM) system has no scram functions. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR less than the fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective:

To assure the availability of an independent reactivity control mechanism.

Specification:

A. Normal Operation

Except as specified ir 3.4.B below, the Standby Liquid Control System shall be operable during periods when fuel is in the reactor unless:

1. The reactor is in cold shutdown

and

2. Control rods are fully inserted and Specification 3.3.A is met.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective:

To verify the operability of the Standby Liquid Control System.

Specification:

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1275 psig shall be verified for each pump by recirculating demineralized water to the test tank.

2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

3. Testing that the setting of the pressure relief valves is between 1400 and 1490 psig.

4. Initiating one of the standby liquid control loops, excluding the primer chamber and inlet fitting, and verifying that a flow path from a pump to the reactor

3.4 LIMITING CONDITIONS FOR OPERATION

4.4 SURVEILLANCE REQUIREMENTS

vessel is available by pumping demineralized water into the reactor vessel. Both loops shall be tested over the course of two operating cycles.

5. Testing the new trigger assemblies by installing one of the assemblies in the test block and firing it using the installed circuitry. Install the unfired assemblies, taken from the same batch as the fired one, into the explosion valves.

6. Recirculating the borated solution.

3.4 B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

4.4 B. Operation with Inoperable Components

When a component becomes inoperable, its redundant component shall be or shall have been demonstrated to be operable within 24 hours.

3.4 C. Liquid Poison Tank - Boron Concentration

At all times when the Standby Liquid Control System is required to be operable, the following conditions shall be met:

1. The net volume versus concentration of the sodium pentaborate solution in the standby liquid control tank shall meet the requirements of Figure 3.4.1.

4.4 C. Liquid Poison Tank - Boron Concentration

1. The solution volume in the tank and temperature in the tank and suction piping shall be checked at least daily.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that the Alternate Cooling Tower Subsystem or both Station Service Water Subsystems are made or found inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem(s) are made operable, provided that during such seven days all other active components of the other subsystem(s) are operable.

4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

E. High Pressure Cooling Injection (HPCI) System

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 150 psig and prior to reactor startup from a cold condition:

   a. The HPCI System shall be operable.

   b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

4.5 SURVEILLANCE REQUIREMENT

3. When the Alternate Cooling Subsystem or both Station Service Water Subsystems are made or found to be inoperable, the operable subsystem(s) shall have been or shall be demonstrated to be operable within 24 hours.

E. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI System shall be performed as follows:

1. Testing

Item	Frequency
Simulated Automatic Actuation Test	Each refueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Depressurization Subsystems, the Core Spray Subsystems, the LPCI Subsystems, and the RCIC System are operable.

3. If the requirements of Specification 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to less than 120 psig within 24 hours.

F. <u>Automatic Depressurization System</u>

1. Except as specified in Specification 3.5.F.2 below, the entire Automatic Depressurization Relief System shall be operable at any time the reactor pressure is above 100 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the Automatic Depressurization Subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the

4.5 SURVEILLANCE REQUIREMENT

2. When the HPCI Subsystem is made or found to be inoperable, the Automatic Depressurization System shall have been or shall be demonstrated to be operable within 24 hours.

NOTE: Automatic Depressurization System operability shall be demonstrated by performing a functional test of the trip system logic.

F. <u>Automatic Depressurization System</u>

Surveillance of the Automatic Depressurization System shall be performed as follows:

1. Operability testing of the relief valves shall be in accordance with Specification 4.6.E.

2. When one relief valve of the Automatic Pressure Relief Subsystem is made or found to be inoperable, the HPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding seven days unless such a valve is sooner made operable, provided that during such seven days both the remaining Automatic Relief System valves and the HPCI System are operable.

3. If the requirements of Specification 3.5.F cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to less than 100 psig within 24 hours.

G. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in Specification 3.5.G.2 below, the RCIC System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that the RCIC System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the HPCI System are operable.

4.5 SURVEILLANCE REQUIREMENT

G. Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Testing

Item	Frequency
Simulated automatic actuation test (testing valve operability)	Each refueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The RCIC System shall deliver at least 400 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION	4.5 SURVEILLANCE REQUIREMENT

3. If the requirements of Specification 3.5.G cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to less than 120 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the emergency diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the LPCI and CS and Containment Cooling Subsystems connecting to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the Core and Containment Cooling Systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all Core and Containment Cooling Subsystems may be inoperable provided no work is permitted which has the potential for draining the reactor vessel.

H. Minimum Core and Containment Cooling System Availability

1. When one of the emergency diesel generators is made or found to be inoperable, the remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.

3.6 LIMITING CONDITIONS FOR OPERATION	4.6 SURVEILLANCE REQUIREMENTS

e. With the radioiodine concentration in the reactor coolant greater than 1.1 microcuries/ gram dose equivalent I-131, a sample of reactor coolant shall be taken every 4 hours and analyzed for radioactive iodines of I-131 through I-135, until the specific activity of the reactor coolant is restored below 1.1 microcuries/ gram dose equivalent I-131.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in Specification 3.6.B.3:

Conductivity	5umho/cm
Chloride ion	0.1 ppm

3. For reactor startups the maximum value for conductivity shall not exceed 10 umho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, in the reactor coolant water for the first 24 hours after placing the reactor in the power operating condition.

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes), and analyzed for conductivity and chloride ion content.

3.6 LIMITING CONDITIONS FOR OPERATION

4. Except as specified in Specification 3.6.B.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hours.

Conductivity	5 uhmo/cm
Chloride ion	0.5 ppm

5. If Specification 3.6.B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

C. Coolant Leakage

1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

   b. While in the run mode, reactor coolant leakage into the primary containment from unidentified sources shall not

4.6 SURVEILLANCE REQUIREMENTS

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample shall be taken every four hours and analyzed for conductivity and chloride ion content.

C. Coolant Leakage

1. Reactor coolant system leakage, for the purpose of satisfying Specification 3.6.C.1, shall be checked and logged once per shift, not to exceed 12 hours.

BASES: 3.6 and 4.6 (Cont'd)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. In order to estimate the material properties at the 1/4 and 3/4 T positions in the vessel plate, the shift in RT_{NDT} is determined in accordance with Regulatory Guide 1.99, Revision 2. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines, shown on Figure 3.6.1, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10CFR Part 50.

B. Coolant Chemistry

A steady-state radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in Specification 3.8.E.1, or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was calculated on the basis of the radioiodine concentration limit of 1.1 µCi of I-131 dose equivalent per gram of water, atmospheric diffusion from an equivalent elevated release of 10 meters at the nearest site boundary (190 m) for a $X/Q = 3.9 \times 10^{-3}$ sec/m^3 (Pasquill D and 0.33 m/sec equivalent), and a steam line isolation valve closure time of five seconds with a steam/water mass release of 30,000 pounds.

The iodine spike limit of four (4) microcuries of I-131 dose equivalent per gram of water provides an iodine peak or spike limit for the reactor coolant concentration to assure that the radiological consequences of a postulated LOCA are within 10CFR Part 100 dose guidelines.

The reactor coolant sample will be used to assure that the limit of Specification 3.6.B.1 is not exceeded. The radioiodine concentration would not be expected to change rapidly during steady-state operation over a period of 96 hours. In addition, the trend of the radioactive gaseous effluents, which is continuously monitored, is a good indicator of the trend of the radioiodine concentration in the reactor coolant. When a significant increase in radioactive gaseous effluents is indicated, as specified, an additional reactor coolant sample shall be taken and analyzed for radioactive iodine.

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.1.b may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shutdown to allow further investigation and corrective action.

The 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

The removal capacity from the drywell floor drain sump and the equivalent drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Parametric evaluations have shown that only three of the four relief valves are required to provide a pressure margin greater than the recommended 25 psi below the safety valve actuation settings as well as a MCPR > 1.06 for the limiting overpressure transient below 98% power. Consequently, 95% power has been selected as a limiting power level for three valve operation. For the purposes of this limiting condition a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

BASES: 3.6 and 4.6 (Cont'd)

E. Structural Integrity and Operability Testing

A pre-service inspection of the components listed in Table 4.2-3 of the FSAR was conducted after site erection to assure freedom from defects greater than code allowance; in addition, this serves as a reference base for further inspections. Prior to operation, the reactor primary system was free of gross defects. In addition, the facility has been designed such that gross defects should not occur

TABLE 4.7.2.a

PRIMARY CONTAINMENT ISOLATION VALVES

VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (1)	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A, D & 2-86A, D)	4	4	5 (Note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SC
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9)		1	10	Closed	SC
3	Drywell Air Purge Inlet (16-19-8)		1	10	Open	GC
3	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed*	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed*	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Open	GC

* Valves 16-19-7 and 16-19-7A shall have stops installed to limit valve opening to 50° or less.

BASES: 4.7 (Cont'd)

The maximum allowable test leak rate at the peak accident pressure of 44 psig (La) is 0.80 weight % per day. The maximum allowable test leak rate at the retest pressure of 24 psig (Lt) has been conservatively determined to be 0.59 weight percent per day. This value was verified to be conservative by actual primary containment leak rate measurements at both 44 psig and 24 psig upon completion of the containment structure.

To allow a margin for possible leakage deterioration between test intervals, the maximum allowable operational leak rate (Ltm), which will be met to remain on the normal test schedule, is 0.75 Lt.

As most leakage and deterioration of integrity is expected to occur through penetrations, especially those with resilient seals, a periodic leak rate test program of such penetration is conducted at the peak accident pressure of 44 psig to insure not only that the leakage remains acceptably low but also that the sealing materials can withstand the accident pressure.

The leak rate testing program is based on AEC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression Chamber-Reactor Building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. Operability testing is performed in conjunction with Specification 4.6.E. Inspections and calibrations are performed during the refueling outages; this frequency being based on equipment quality, experience, and engineering judgment.

The ten (10) drywell-suppression vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified. Also it is inspected to assure that it closes freely and operates properly.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.12 ft^2. This is equivalent to one vacuum breaker open by three-eighths of an inch (3/8") as measured at all points around the circumference of the disk or three-fourths of an inch (3/4") as measured at the bottom of the disk when the top of the disk is on the seat. Since these valves open in a manner that is purely neither mode, a conservative allowance of one-half inch (1/2") has been selected as the maximum permissible valve opening. Assuming that permissible valve opening could be evenly divided among all ten vacuum breakers at once, valve open position assumed to indication for an individual valve must be activated less than fifty-thousandths of an inch (0.050") at all points along the seal surface of the disk. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a nonseated valve.

BASES: 3.8 (Cont'd)

liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10CFR Part 50, for liquid effluents.

D. Liquid Holdup Tanks

The tanks listed in this Specification include all outdoor tanks that contain radioactivity that are not surrounded by liners, dikes, or walls capable of holding the tank contents, or that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

E. Gaseous Effluents: Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents will be within the annual dose limits of 10CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of member(s) of the public either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10CFR Part 20 [10 CFR Part 20.106(b)]. For member(s) of the public who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified limits as determined by the methodology in the ODCM, restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to (500) mrem/year to the total body or to (3,000) mrem/year to the skin.

Specification 3.8.E.b also restricts, at all times, comparable with the length of the sampling periods of Table 4.8.2 the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to 1500 mrem/year for the nearest cow to the plant.

F. Gaseous Effluents: Dose from Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A, and IV.A of Appendix I, 10CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The requirements provide operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I, i.e., that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of any member of the public through appropriate pathways is unlikely to be substantially underestimated. The appropriate dose equations are

TABLE 3.9.1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	Minimum Channels Operable	Notes
1. Gross Radioactivity Monitors not Providing Automatic Termination of Release		
a. Liquid Radwaste Discharge Monitor (RD-17-330/PP-17-331/RM-17-350)	1*	1,4,5
b. Service Water Discharge Monitor (RD-17-332/PP-17-333/RM-17-351)	1	2,4,5
2. Flow Rate Measurement Devices		
a. Liquid Radwaste Discharge Flow Rate Monitor (FIT-20-485/442/FR-20-441)	1*	3,4

* During releases via this pathway.

TABLE 3.9.1 NOTATION

NOTE 1 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.8.A.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

NOTE 2 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

NOTE 3 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

NOTE 4 - With the number of channels operable less than required by the minimum channels operable requirement, exert reasonable efforts to return the instrument(s) to operable status prior to the next release.

NOTE 5 - The alarm setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Off-Site Dose Calculation Manual (ODCM). With a radioactive liquid effluent monitoring instrumentation channel alarm setpoint less conservative than a value which will ensure that the limits of 3.8.A.1 are met during periods of release, immediately take action to suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable; or change the setpoint so it is acceptably conservative.

TABLE 3.9.2

GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels Operable	Notes
1. Steam Jet Air Ejector (SJAE)		
a. Noble Gas Activity Monitor (RD-17-130A/RM-17-150A, RD-17-130B/RM-17-150B)	1	7, 8, 9
2. Augmented Off-Gas System		
a. Noble Gas Activity Monitor Between the Charcoal Bed System and the Plant Stack (Providing Alarm and Automatic Termination of Release) (RE OG-3107/RAN OG-3127, RE OG-3108/RAN OG-3128)	1	2, 5, 6, 7
b. Flow Rate Monitor (FE OG-1802/FT OG-1902/FI-OG-2002, FE OG-1804/FT OG-1904/FI-OG-2004, FE OG-1805/FT OG-1905/FI-OG-2008)	1	1, 5, 6
c. Hydrogen Monitor (H2E OG-2901A/H2AN OG-2921A, H2E OG-2901B/H2AN OG-2921B, H2E OG-2902A/H2AN OG-2922A, H2E OG-2902B/H2AN OG-2922B)	1	3, 5, 6
3. Plant Stack		
a. Noble Gas Activity Monitor (RD/RM-17-156, RD/RM-17-157)	1	5, 7, 10
b. Iodine Sampler Cartridge	1	4, 5
c. Particulate Sampler Filter	1	4, 5
d. Sampler Flow Integrator (FI-17-156/157)	1	1, 5
e. Stack Flow Rate Monitor (FE-108-22A&B/FA-108-22/FI-108-22)	1	1, 5

TABLE 3.9.3
(Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. DIRECT RADIATION[b]	40 routine monitoring stations as follows: 16 incident response stations (one in each meteorological sector) within a range of 0 to 4 km[g]; 16 incident response stations (one in each meteorological sector) within a range of 2 to 8 km[g]; the balance of the stations to be placed in special interest areas and control station areas.	Quarterly.	Gamma dose, at least once per quarter. Incident response TLDs in the outer monitoring locations, de-dose only quarterly unless gaseous release LCO was exceeded in period.

TABLE 3.9.4

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES[a]

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Vegetation (pCi/Kg, wet)	Sediment (pCi/Kg, dry)
H-3	2×10^4[b]					
Mn-54	1×10^3		3×10^4			
Fe-59	4×10^2		1×10^4			
Co-58	1×10^3		3×10^4			
Co-60	3×10^2		1×10^4			3×10^3[c]
Zn-65	3×10^2		2×10^4			
Zr-Nb-95	4×10^2					
I-131		0.9		3	1×10^2	
Cs-134	30	10	1×10^3	60	1×10^3	
Cs-137	50	20	2×10^3	70	2×10^3	
Ba-La-140	2×10^2			3×10^2		

(a) Reporting levels may be averaged over a calendar quarter. When more than one of the radionuclides in Table 3.9.4 are detected in the sampling medium, the unique reporting requirements are not exercised if the following condition holds:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots < 1.0.$$

When radionuclides other than those in Table 3.9.4 are detected and are the result of plant effluents, the potential annual dose to a member of the public must be less than or equal to the calendar year limits of Specifications 3.8.B, 3.8.E and 3.8.F.

(b) Reporting level for drinking water pathways. For nondrinking water pathways, a value of 3×10^4 pCi/l may be used.

(c) Reporting level for individual grab samples taken at North Storm Drain Outfall only.

TABLE 4.9.2 NOTATION

(1) The Instrument Functional Test shall also demonstrate that automatic isolation of this pathway and the Control Room alarm annunciation occurs if any of the following conditions exists:

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls not set in operate mode.

(2) The Instrument Functional Test shall also demonstrate that Control Room alarm annunciation occurs when any of the following conditions exist:

(a) Instrument indicates measured levels above the alarm setpoint.

(b) Circuit failure.

(c) Instrument indicates a downscale failure.

(d) Instrument controls are not set in operate mode.

(3) The Instrument Calibration for radioactivity measurement instrumentation shall include the use of a known (traceable to National Bureau of Standards) radioactive source positioned in a reproducible geometry with respect to the sensor. These standards should permit calibrating the system over its normal operating range of rate capabilities.

(4) The Instrument Calibration shall include the use of standard gas samples (high range and low range) containing suitable concentrations, hydrogen balance air, for the detection range of interest per Specification 3.8.J.1.

BASES:

3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

A. Liquid Effluent Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm setpoints for these instruments are to ensure that the alarm will occur prior to exceeding the limits of 10CFR Part 20.

Automatic isolation function is not provided on the liquid radwaste discharge line due to the infrequent nature of batch, discrete volume, liquid discharges (on the order of once per year or less), and the administrative controls provided to ensure that conservative discharge flow rates/dilution flows are set such that the probability of exceeding the 10CFR Part 20 concentration limits are low, and the potential off-site dose consequences are also low.

B. Gaseous Effluent Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments are provided to ensure that the alarm/trip will occur prior to exceeding the limits of 10CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

C. Radiological Environmental Monitoring Program

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of member(s) of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

Ten years of plant operation, including the years prior to the implementation of the Augmented Off-Gas System, have amply demonstrated via routine effluent and environmental reports that plant effluent measurements and modeling of environmental pathways are adequately conservative. In all cases, environmental sample results have been two to three orders of magnitude less than expected by the model employed, thereby representing small percentages of the ALARA and environmental reporting levels. This radiological environmental monitoring program has therefore been significantly modified as provided for by Regulatory Guide 4.1 (C.2.b), Revision 1, April 1975. Specifically, the air particulate and radioiodine air sampling periods have been increased to semimonthly, based on plant effluent and environmental air sampling data for the previous ten years of operation. An I-131 release rate trigger value of 1×10^{-1} uCi/sec from the plant stack will require that air sample collection be increased to weekly. The

3.12 LIMITING CONDITIONS FOR OPERATION

E. Extended Core Maintenance

More than two control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.12.B.

3. If the spiral unload/reload method of core alteration is to be used, the following conditions shall be met:

   a. Prior to spiral unload and reload, the SRMs shall be proven operable as stated in Specification 3.12.B1 and 3.12.B2. However, during spiral unloading, the count rate may drop below 3 cps.

4.12 SURVEILLANCE REQUIREMENTS

E. Extended Core Maintenance

Prior to control rod withdrawal for extended core maintenance, that control rod's control cell shall be verified to contain no fuel assemblies.

1. This surveillance requirement is the same as that given in Specification 4.12.A.

2. This surveillance requirement is the same as that given in Specification 4.12.B.

BASES:

3.12 & 4.12 REFUELING

A. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the Reactivity Control System and interlocks at any time during shutdown or the following operating cycle.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist.

Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position, only one control rod can be withdrawn.

B. The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored. Under the special condition of complete spiral core unloading, it is expected that the count rate of the SRMs will drop below 3 cps before all the fuel is unloaded. Since there will be no reactivity additions, a lower number of counts will not present a hazard. When all of the fuel has been removed to the spent fuel storage pool, the SRMs will no longer be required. Requiring the SRMs to be operational prior to fuel removal assures that the SRMs are operable and can be relied on even when the count rate may go below 3 cps.

Prior to spiral reload, two diagonally adjacent fuel assemblies, which have previously accumulated exposure in the reactor, will be loaded into their designated core positions next to each of the 4 SRMs to obtain the required 3 cps. Exposed fuel continuously produces neutrons by spontaneous fission of certain plutonium isotopes, photo fission, and photo disintegration of deuterium in the moderator. This neutron production is normally great enough to meet the 3 cps minimum SRM requirement, thereby providing a means by which SRM response may be demonstrated before the spiral reload begins. During the spiral reload, the fuel will be loaded in the reverse sequence that it was unloaded with the exception of the initial eight (8) fuel assemblies which are loaded next to the SRMs to provide a means of SRM response.

3.13 LIMITING CONDITIONS FOR OPERATION

C. Fire Hose Stations

1. Except as specified in 3.13.C.2 below, all hose stations inside the Reactor Building, Turbine Building, and those inside the Administration Building which provided coverage of the Control Room Building shall be operable whenever equipment in the areas protected by the fire hose stations is required to be operable.

2. With one or more of the fire hose stations specified in 3.13.C.1 above inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an operable hose station within one hour.

4.13 SURVEILLANCE REQUIREMENTS

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.

C. Fire Hose Stations

1. Each fire hose station shall be verified to be operable:

a. At least monthly by visual inspection of the station to assure all equipment is available.

b. At least once each 18 months by removing the hose for inspection and replacing degraded coupling gaskets and reracking.

c. At least once each year by hydro-statically testing each outside hose at 250 lbs.

d. At least once per 3 years by hydro-statically testing inside hose at 150 lbs.

3.13 LIMITING CONDITIONS FOR OPERATION

D. High Pressure CO_2 Systems

1. Except as specified in Specification 3.13.D.2, the CO_2 systems located in the cable vault, switchgear rooms, and diesel fire pump day tank room shall be operable, whenever equipment in the area protected by the system is required to be operable.

2. From and after the date that the CO_2 system in the cable vault or a switchgear room is inoperable, within one hour a fire watch shall be established to inspect the location at least once every hour, provided that the fire detection system is operable in accordance with 3.13.A. If the fire detection system is also inoperable, within one hour a continuous fire watch shall be established with backup fire suppression equipment. Restore the CO_2 system to operable status within 14 days or submit a report within the next 30 days to the Commission as specified in 6.7.C.2 outlining the cause of inoperability and the plans for restoring the CO_2 system to operable status.

4.13 SURVEILLANCE REQUIREMENTS

e. At least once per 3 years, partially open hose station valves to verify valve operability and no blockage.

D. High Pressure CO_2 Systems

1. The CO_2 systems located in the cable vault, switchgear rooms, and diesel fire pump day tank room shall be demonstrated operable.

a. At least once per six months by verifying each CO_2 cylinder does not contain less than 90% of its initial charge.

b. At least once per 18 months by verifying that the system, including associated ventilation dampers, will actuate automatically to a simulated actuation signal.

c. At least once per operating cycle a flow path test shall be performed to verify flow through each nozzle.

3.13 LIMITING CONDITIONS FOR OPERATION	4.13 SURVEILLANCE REQUIREMENTS
	c. At least once per 3 years by performing an air flow test through the Recirculation M.G. Set foam header and verifying each foam nozzle is unobstructed.

TABLE 3.13.A.1

FIRE DETECTION SENSORS

	Sensor Location	Minimum No. of Sensors Required to Be Operable: Heat	Flame	Smoke
1.	Cable Spreading Room & Station Battery Room	-	-	23
2.	Switchgear Room (East)	-	-	10
3.	Switchgear Room (West)	-	-	10
4.	Diesel Generator Room (A)	-	-	2
5.	Diesel Generator Room (B)	-	-	2
6.	Intake Structure (Service Water)	1	1	1
7.	Recirc Motor Generator Set Area	3	-	8
8.a	Control Room Zone 1 (Control Room Ceiling)	-	-	14
8.b	Control Room Zone 2 (Control Room Panels)	-	-	18
8.c	Control Room Zone 3 (Control Room Panels)	-	-	25
8.d	Control Room Zone 4 (Control Room Panels)	-	-	10
8.e	Control Room Zone 5 (Exhaust & Supply Ducts)	-	-	2
9.a	Rx Bldg. Corner Rm NW 232	-	-	1
9.b	Rx Bldg. Corner Rm NW 213 (RCIC)	-	-	1
9.c	Rx Bldg. Corner Rm NE 232	-	-	1
9.d	Rx Bldg. Corner Rm NE 213	-	-	1
9.e	Rx Bldg. Corner Rm SE 232	-	-	1
9.f	Rx Bldg. Corner Rm SE 213	-	-	1
9.g	Rx Bldg. Corner Rm SW 232	-	-	1
10.	HPCI Room	-	-	8
11.	Torus area	12	-	16
12.	Rx Bldg. Cable Penetration Area	-	-	7
13.	Refuel Floor	-	-	13
14.	Diesel Oil Day Tank Room (A)	-	1*	1*
15.	Diesel Oil Day Tank Room (B)	-	1*	1*
16.	Turbine Loading Bay (vehicles)	-	3	-

*NOTE: The Diesel Day Tank Rooms require only one detector operable (1 flame or 1 smoke).

6.0 ADMINISTRATIVE CONTROLS

Administrative controls are the written rules, orders, instructions, procedures, policies, practices, and the designation of authorities and responsibilities by the management to obtain assurance of safety and quality of operation and maintenance of a nuclear power reactor. These controls shall be adhered to.

6.1 ORGANIZATION

A. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organizational positions. These relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the Yankee Operational Quality Assurance Manual.

B. The Plant Manager shall be responsible for overall unit safe operation and shall have control over those on-site activities necessary for safe operation and maintenance of the plant. Succession to this responsibility during his absence shall be delegated in writing.

C. The Manager of Operations 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.

D. Conduct of operations of the plant will be in accordance with the following minimum conditions.

1. An individual qualified in radiation protection procedures shall be present on-site at all times when there is fuel in the reactor.

2. Minimum shift staffing on-site shall be in accordance with Table 6.1.1.

3. A dedicated, licensed Senior Operator shall be in charge of any reactor core alteration.

4. Qualifications with regard to educational background experience, and technical specialties of the key supervisory personnel listed below shall apply and be maintained in accordance with the levels described in the American National Standards Institute N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants".

   a. Plant Manager
   b. Superintendent(s)
   c. Chemistry Manager
   d. Radiation Protection Manager
   e. Operations Manager (See Item 6.1.D.7)
   f. Reactor Engineering Manager
   g. Maintenance Manager
   h. Instrument and Control Manager
   i. Shift Supervisors

6.2 REVIEW AND AUDIT

Organizational units for the review and audit of plant operations shall be constituted and have the responsibilities and authorities outlined below:

A. Plant Operations Review Committee

1. Membership

   a. Chairman: Plant Manager
   b. Vice-Chairman: Superintendent(s)
   c. Engineering Manager
   d. Operations Manager
   e. Maintenance Manager
   f. Reactor Engineering Manager
   g. Chemistry Manager
   h. Instrument and Control Manager
   i. Radiation Protection Manager

2. Qualifications

   The qualifications of the regular members of the Plant Operations Review Committee with regard to the combined experience and technical specialties of the individual members shall be maintained at a level at least equal to or higher than as described in Specification 6.1.

3. Meeting Frequency: Monthly, and as required, on call of the Chairman.

4. Quorum: Chairman or Vice-Chairman plus four members or their designated alternates.

   NOTE: For purposes of satisfying a quorum, a Vice-Chairman may be considered a member providing that Vice-Chairman is not presiding over the meeting.

5. Designated alternates shall be from other plant personnel in the appropriate disciplines or as selected by the Plant Manager; however, there shall be no more than three (3) alternates serving on the committee at any one time.

6. Responsibilities

   a. Review proposed normal, abnormal, and emergency operating procedures. Review all proposed maintenance procedures and proposed changes to those procedures; and any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

   b. Review proposed tests and experiments.

   c. Review proposed changes to Technical Specifications.

   d. Review proposed changes or modifications to plant systems or equipment, which changes would require a change in procedures in (a) above.

   e. Review plant operations to detect any potential safety hazards.

f. The Committee membership and its Chairman and Vice Chairman shall be appointed as specified in the Yankee Quality Assurance Manual.

2. Qualifications

The Committee shall consist of a minimum of six (6) members plus designated alternates who as a group employ expertise in the following areas:

a. Nuclear Power Plant Technology

b. Reactor Operations

c. Utility Operations

d. Power Plant Design

e. Reactor Engineering

f. Radiation Safety

g. Safety Analysis

h. Instrumentation and Control

i. Metallurgy

3. Meeting Frequency: Semi-annually and as required on call of the Chairman.

4. Quorum: Chairman or Vice Chairman plus four members or designated alternates.

5. Responsibilities:

a. Review proposed changes to the operating license including Technical Specifications.

b. Review minutes of meetings of the Plant Operation Review Committee to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

c. Review the safety evaluations for changes to equipment or systems completed under the provisions of Section 50.59 10 CFR to verify that such actions did not constitute an unreviewed safety question.

d. Periodic audits of implementing procedures, shall be performed under cognizance of the Committee. Included in these audits, but not limited to, are the following specific activities:

i. plant operations;

ii. facility fire protection program;

iii. the radiological environmental monitoring program and the results thereof at least once per 12 months;

iv. the Off-Site Dose Calculation Manual and implementing procedures at least once per 24 months;

10. Records for Environmental Qualification which are covered under the provisions of paragraph 6.9.

11. Records of analysis required by the Radiological Environmental Monitoring Program.

6.7 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10 Code of Federal Regulations, the following identified reports shall be submitted to the appropriate Regional Office unless otherwise noted.

A. Routine Reports

1. Startup Report

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 plant. The report shall address each of the tests identified in the FSAR and shall, in general, include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption of commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

2. Annual Report

An annual report covering the previous calendar year shall be submitted prior to March 1 of each year. The annual report shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

1/ This tabulation supplements the requirements of 20.407 of 10CFR Part 20.

10. Records for Environmental Qualification which are covered under the provisions of paragraph 6.9.

11. Records of analysis required by the Radiological Environmental Monitoring Program.

6.7 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10 Code of Federal Regulations, the following identified reports shall be submitted to the appropriate Regional Office unless otherwise noted.

A. Routine Reports

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1/ This tabulation supplements the requirements of 20.407 of 10CFR Part 20.

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Supplemental Information to VYNPC April 19, 1990 Response Regarding FROSSTEY-2 Fuel Performance Code," BVY 90-054, dated May 10, 1990 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "Responses to Request for Additional Information on FROSSTEY-2 Fuel Performance Code," BVY 91-024, dated March 6, 1991 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "LOCA-Related Responses to Open Issues on FROSSTEY-2 Fuel Performance Code," BVY 92-39, dated March 27, 1992 (Approved by NRC SER, dated September 24, 1992).

Letter from L. A. Tremblay, Jr. (VYNPC) to USNRC, "FROSSTEY-2 Fuel Performance Code - Vermont Yankee Response to Remaining Concerns," BVY 92-54, dated May 15, 1992 (Approved by NRC SER, dated September 24, 1992).

Report, "Loss-of-Coolant Accident Analysis for Vermont Yankee Nuclear Power Station," NEDO-21697, August 1977, as amended (Approved by NRC SER, dated November 30, 1977).

Report, "General Electric Standard Application for Reactor Fuel (GESTARII)," NEDE-24011-P-A, GE Company Proprietary (the latest NRC-approved version will be listed in the COLR).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Reportable Occurrences

This section deleted.

C. Unique Reporting Requirements

1. Annual Radioactive Effluent Release Report

a. Within 90 days after January 1 of each year, a report shall be submitted covering the radioactive content of effluents released to unrestricted areas during the previous calendar year of operation.

6.13 OFF-SITE DOSE CALCULATION MANUAL (ODCM)

An Off-Site Dose Calculation Manual shall contain the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

A. Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Annual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM which were changed with each page numbered and provided with the revision number, together with appropriate analyses or evaluations justifying the change(s).

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

c. Documentation of the fact that the change has been reviewed by PORC and approved by the Manager of Operations (MOO).

2. Shall become effective upon review by PORC and approved by the Manager of Operations (MOO).

6.14 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS, AND SOLID WASTE TREATMENT SYSTEMS*

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

A. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PORC. The discussion of each change shall contain:

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

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

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

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* Licensee may choose to submit the information called for in this Specification as part of the periodic FSAR update.