ADMINISTRATIVE CONTROLS

SECTION		PAGE
6.1 RESPONSIBILITY		6-1
6.2 ORGANIZATION		6-1
6.2.1	Corporate	6-1
	~~Figure 6.2.1-1 Corporate Organization~~	~~6-5~~
6.2.2	Unit Staff	6-1
	~~Figure 6.2.2-1 Unit Organization~~	~~6-4~~
	Table 6.2.2-1 Minimum Shift Crew Composition	6-6
6.2.3	INDEPENDENT SAFETY ENGINEERING GROUP	
	Function	6-7
	Composition	6-7
	Responsibilities	6-7
	Records	6-7
6.2.4	SHIFT TECHNICAL ADVISOR	6-7
6.3 UNIT STAFF QUALIFICATIONS		6-7
6.4 TRAINING		6-8
6.5 REVIEW AND AUDIT		
6.5.1	PLANT OPERATIONS REVIEW COMMITTEE (PORC)	
	Function	6-8
	Composition	6-8
	Alternates	6-8
	Meeting Frequency	6-8
	Quorum	6-9
	Responsibilities	6-9
	Records	6-10

9009280145 900913
PDR ADOCK 05000440
P PNU

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
ACTION

INSERT 1 HERE →

ACTION 20 - ~~Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.~~

ACTION 21 - Close the affected system isolation valve(s) within one hour or:
a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or:
a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
b. In OPERATIONAL CONDITION *, suspend CORE ALTERATIONS, operations with a potential for draining the reactor vessel, and handling of irradiated fuel in the primary containment.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Verify SECONDARY CONTAINMENT INTEGRITY with the annulus exhaust gas treatment system operating within one hour.##

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within 1 hour and declare the affected system inoperable.##

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.##

ACTION 28 - Within one hour lock the affected system isolation valves closed, or verify, by remote indication, that the valve(s) is closed and electrically disarmed, or isolate the penetration(s) and declare the affected system inoperable.

NOTES

* When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or the key locked Condenser Low Vacuum Bypass Switch is in the normal position.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

The provisions of Specification 3.0.4 are not applicable.

Insert 1

Action 20 - In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours. In OPERATIONAL CONDITION #, suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

NJC/CODED/3761

INSTRUMENTATION

TRAVERSING IN-CORE PROBE SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.7.7 The traversing in-core probe system shall be OPERABLE with either:

a. Five movable detectors, drives and readout equipment to map the core, and indexing equipment to allow all five detectors to be calibrated in a common location.

OR

b. With one or more TIP measurement locations inoperable, data may be replaced by data obtained from that location's symmetric counterpart if the substitute TIP data was obtained from an OPERABLE measurement location; provided the reactor core is operating in a type A control rod pattern and the total core TIP uncertainty for the present cycle has been determined to be less than 0.7 percent (standard deviation). Symmetric counterpart data may be substituted for a maximum of ten TIP measurement locations.

APPLICABILITY: When the traversing in-core probe is used for:

a.* Recalibration of the LPRM detectors, and

b.* Monitoring the APLHGR, LHGR, ~~or~~ MCPR, ~~or MFLPD.~~

ACTION:

With the traversing in-core probe system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.7 The traversing in-core probe system shall be demonstrated OPERABLE by normalizing each of the above required detector outputs within 72 hours prior to use when required for the ~~LPRM calibration function~~ above applicable LPRM calibration and monitoring functions.

*Only the detector(s) in the location(s) of interest are required to be OPERABLE.

DEFINITIONS

DRYWELL INTEGRITY (continued)

f. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

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

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

1.12 $\bar{E}$ shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.13 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.14 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

DELETE →

~~FRACTION OF LIMITING POWER DENSITY~~

~~1.15 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~

~~FRACTION OF RATED THERMAL POWER~~

~~1.16 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.~~

DEFINITIONS

SECTION

1.0 DEFINITIONS	PAGE
1.1 ACTION	1-1
1.2 AVERAGE PLANAR EXPOSURE	1-1
1.3 AVERAGE PLANAR LINEAR HEAT GENERATION RATE	1-1
1.4 CHANNEL CALIBRATION	1-1
1.5 CHANNEL CHECK	1-1
1.6 CHANNEL FUNCTIONAL TEST	1-1
1.7 CORE ALTERATION	1-2
1.8 CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY	1-2
1.9 CRITICAL POWER RATIO	1-2
1.10 DOSE EQUIVALENT I-131	1-2
1.11 DRYWELL INTEGRITY	1-2
1.12 E-AVERAGE DISINTEGRATION ENERGY	1-3
1.13 EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	1-3
1.14 END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME	1-3
~~1.15 FRACTION OF LIMITING POWER DENSITY~~	~~1-3~~
~~1.16 FRACTION OF RATED THERMAL POWER~~	~~1-3~~
1.17 FREQUENCY NOTATION	1-4
1.18 FUEL HANDLING BUILDING INTEGRITY	1-4
1.19 GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM	1-4
1.20 IDENTIFIED LEAKAGE	1-4
1.21 ISOLATION SYSTEM RESPONSE TIME	1-4
1.22 LIMITING CONTROL ROD PATTERN	1-5
1.23 LINEAR HEAT GENERATION RATE	1-5

INSTRUMENTATION

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by performance of a CHANNEL CALIBRATION of the turbine overspeed protection instrumentation.

DELETE (*)

c. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of all valve seats, disks and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

~~*The initial surveillance test interval is extended on a one time basis to the first refueling outage.~~

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F*, and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F*, or

DELETE (*)

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F* and the operating loop flow rate is less than or equal to 50% of rated loop flow.

DELETE (*)

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

*Below 25 psig, this temperature differential is not applicable.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a. The formula to be used is: $[L_o + L_{am} - 0.25\ L_a] \leq L_c \leq [L_o + L_{am} + 0.25\ L_a]$ where L_c = supplemental test result; L_o = superimposed leakage; L_{am} = measured Type A leakage.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the primary containment or bled from the primary containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

DELETE (#)

d. Type B and C tests shall be conducted with gas at P_a, 11.31 psig*, at intervals no greater than 24 months except for tests involving:

1. Air locks,

2. Main steam line isolation valves,

3. Valves pressurized with fluid from a seal system,

4. All containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment, and

5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J of 10 CFR 50 Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 12.44 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

DELETE (#)

h. All containment isolation valves in hydrostatically tested lines per Table 3.6.4-1 which penetrate the primary containment shall be leak tested at least once per 18 months.

*Unless a hydrostatic test is required per Table 3.6.4-1.

#A Type C test interval extension to the first refueling outage is permissible for primary containment isolation valves listed in Table 3.6.4-1, which are identified in letter PY-CEI/NRR-0716 L (dated September 11, 1987) as needing a plant outage to test. For this one time test interval, the provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

DRYWELL AND CONTAINMENT PURGE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and containment purge 42-inch outboard (1M14-F040, F090) supply and exhaust isolation valves and the 18-inch supply and exhaust isolation valves (1M14-F190, F195, F200, F205) shall be OPERABLE and:

a. Each 42-inch inboard purge valve (1M14-F045, F085) shall be sealed closed.

b. Each 42-inch outboard purge valve (1M14-F040, F090) may be open limited to an opening angle of 50° or less for purge system operation* with such operation limited to 3000 hours per 365 days for reducing airborne activity and pressure control.

c. Each 24-inch (1M14-F055A, B and F060A, B) and 36-inch (1M14-F065, F070) drywell purge valve shall be sealed closed.

d. Each 2-inch (1M51-F090 and F110) backup hydrogen purge system isolation valves may be open for controlling drywell pressure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a 42-inch inboard drywell and containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, within 4 hours close and/or seal the 42-inch valve(s) or otherwise isolate the penetration or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a 18-inch or 42-inch outboard drywell and containment purge supply and/or exhaust isolation valves inoperable or open for more than 3000 hours per 365 days for purge system operation*, within four hours close the open 18- or 42-inch valve(s) or otherwise isolate the penetration(s) or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With a 24- or 36-inch drywell purge supply and/or exhaust isolation valve(s) open or not sealed closed, within 4 hours close and/or seal close the 24- or 36-inch valve(s) or otherwise isolate the penetration, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With a drywell and containment purge supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.3 and/or

*Purge system operation shall be defined as any time that both 18-inch and the 42-inch outboard purge valves are open concurrently in either the supply or exhaust line.

~~**Applicable from initial fuel load until 3 months following the completion of the first refueling outage; otherwise, a 1000 hour per 365 day limit applies.~~

PLANT SYSTEMS

BASES

SNUBBERS (Continued)

DELETE 10% 1. Functionally test 10% of a type of snubber with an additional ~~10%~~ 5% tested for each functional testing failure, or

INSERT 5% 2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7.4-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.7.4-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be lised in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (i.e., newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.5 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

A.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Three separate and independent diesel generators, each with:

1. A separate day fuel tank containing a minimum of 225 gallons of fuel for Div 1 and Div 2 and 204 gallons of fuel for Div 3,

2. A separate fuel storage system containing a minimum of 69,430 gallons of fuel for Div 1 and Div 2 and 34,824 gallons of fuel for Div 3, and

3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

INSERT → once per

a. With one offsite circuit of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least 8 hours thereafter. If either diesel generator Div 1 or Div 2 has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 for each such diesel generator separately within 24 hours. Restore the offsite circuit to OPERABLE status within 7 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With either diesel generator Div 1 or Div 2 inoperable, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 separately for each diesel generator within 24 hours*;

---

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY. The provisions of Specification 3.0.2 are not applicable.

TABLE 4.8.1.1.2-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	Number of Failures in Last 100 Valid Tests*	Test Frequency
≤ 1	≤ 4	Once per 31 days
≥ 2**	≥ 5	Once per 7 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul# to like-new condition is completed, provided that the overhaul including appropriate post-maintenance operation and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirement ~~4.8.1.1.2.a.4~~ and 4.8.1.1.2.a.5, four tests, in accordance with the 184-day testing requirement of Surveillance Requirements 4.8.1.1.2.a.4 and ~~4.8.1.1.2.a.5~~. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

INSERT
4.8.1.1.2.a.4
4.8.1.1.2.a.5

**The test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to less than or equal to one.

#A one-time waiver to the requirement for performance of a complete diesel generator overhaul to like-new condition has been granted in order to rezero four control air related diesel generator failures (valid failures Nos. 3 through 6 which occurred on 8/11/86, 2/27/87, 3/17/87 and 10/15/87 respectively).

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.4 REMOTE SHUTDOWN INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown monitoring instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. The CHANNEL CHECK for the Primary Containment Isolation Valve Position consists of the verification that indication of valve position (open or closed) can be determined by the valve position lights in the control room. The CHANNEL CALIBRATION for the Primary Containment Isolation Valve Position consists of the Position Indicator Test (PIT), which is conducted in accordance with Specification 4.0.5.

3/4.3.7.6 SOURCE RANGE MONITORS

The source range monitors provide the operator with information of the status of the neutron level in the core at very low power levels during startup and shutdown. At these power levels, reactivity additions shall not be made without this flux level information available to the operator. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

The SRMs are required OPERABLE in OPERATIONAL CONDITION 2 to provide for rod block capability, and are required OPERABLE in ~~OPERABLE~~ CONDITIONS 3 and 4 to provide monitoring capability which provides diversity of protection to the mode switch interlocks.

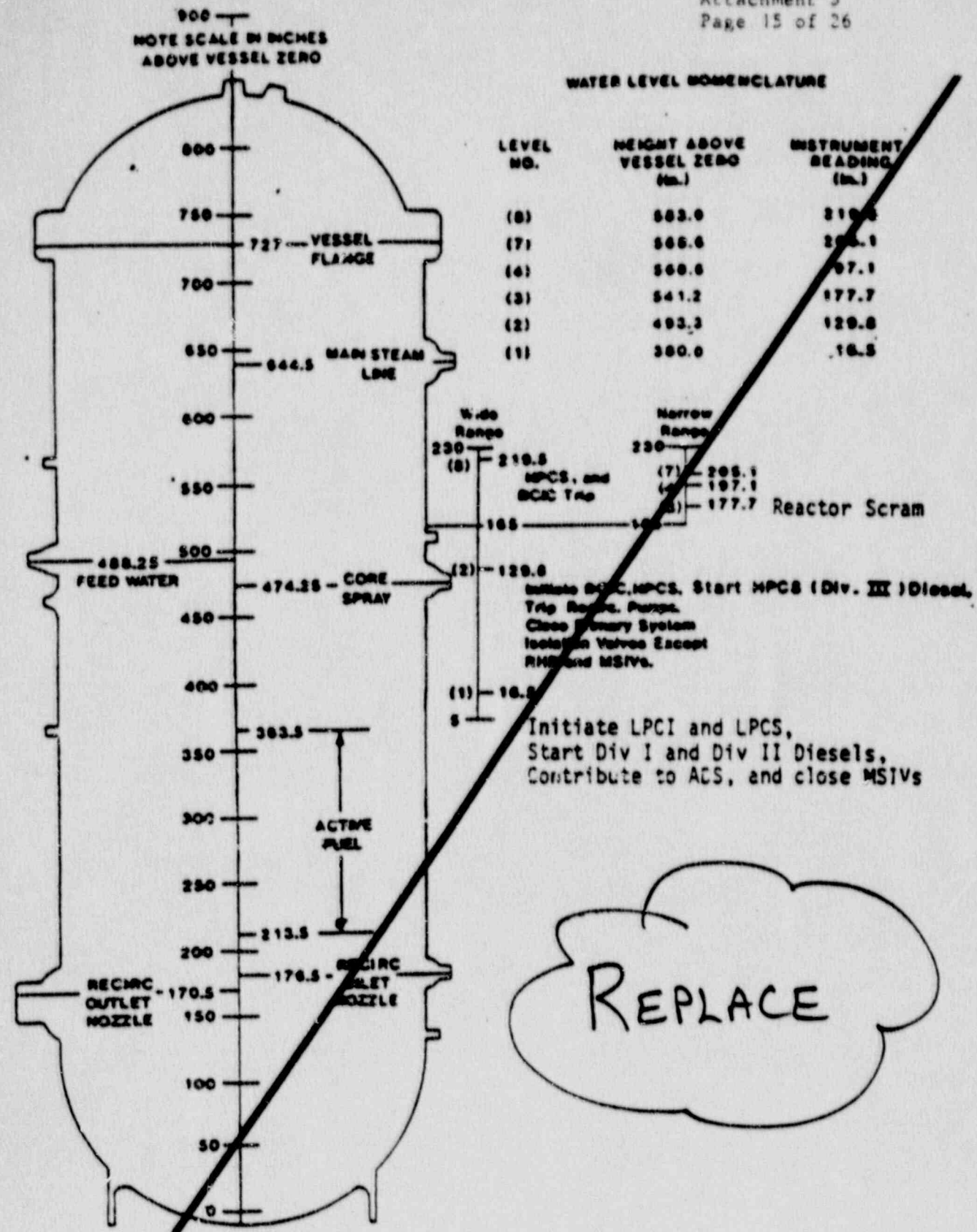
OPERATIONAL

3/4.3.7.7 TRAVERSING IN-CORE PROBE SYSTEM

The OPERABILITY of the traversing in-core probe system with the specified minimum complement of equipment ensures that the measurements obtained from use of this equipment accurately represent the spatial gamma flux distribution of the reactor core. With less than the specified complement of equipment, the spatial gamma flux distribution of the reactor core can still be accurately represented by using replacement data from symmetrical strings (LPRM locations), provided the conditions specified in the LCO are met.

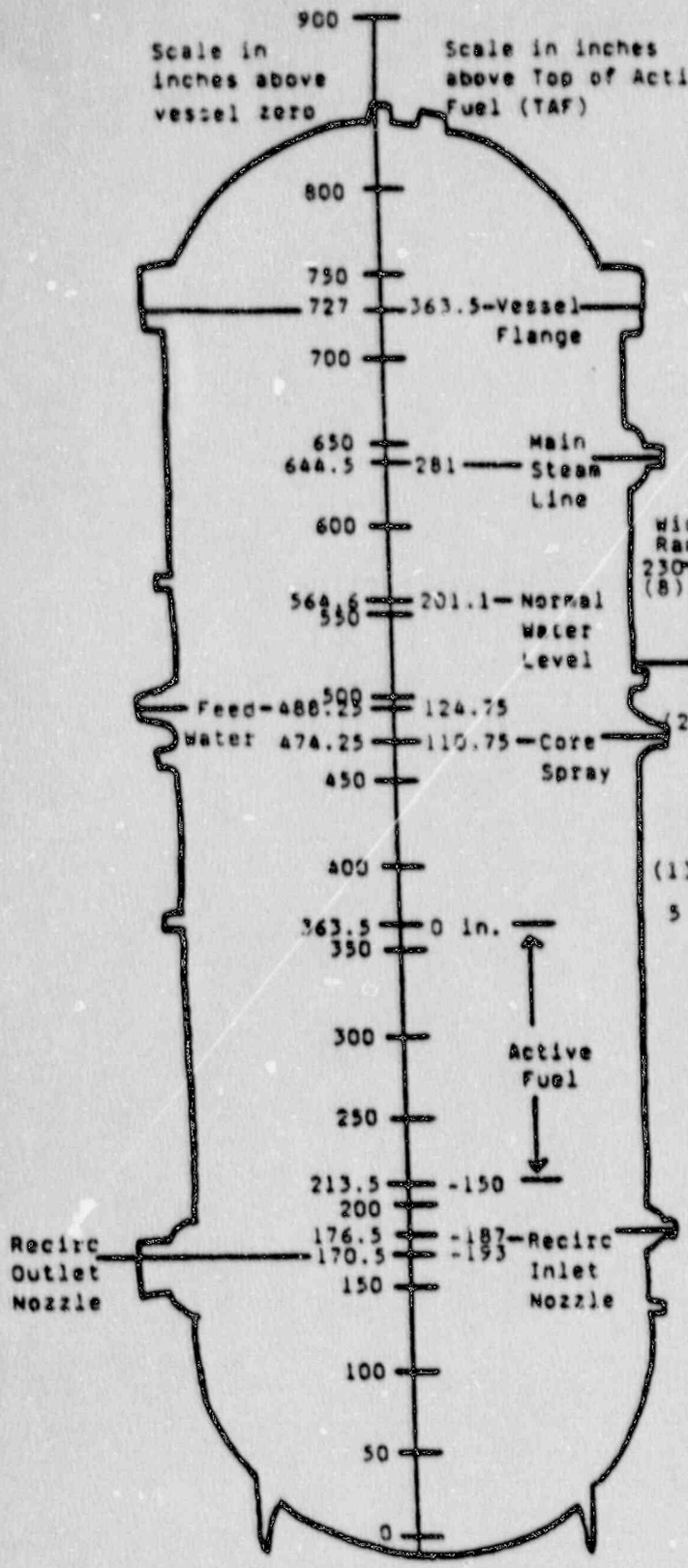
be

The TIP system OPERABILITY is demonstrated by normalizing all probes (i.e., detectors) prior to performing an LPRM calibration function. Monitoring core thermal limits may involve utilizing individual detectors to monitor selected areas of the reactor core, thus all detectors may not be required to OPERABLE. The OPERABILITY of individual detectors to be used for monitoring is demonstrated by comparing the detector(s) output with data obtained during the previous LPRM calibrations.



REACTOR VESSEL WATER LEVEL

Bases Figure B 3/4 3-1



WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	INSTRUMENT ZERO (HEIGHT ABOVE TAF) (INCHES)
(8)	583.0	219.5
(7)	560.6	205.1
(6)	560.6	197.1
(3)	561.2	177.7
(2)	493.3	129.8
(1)	380.0	16.5

Wide Range

230
(8) 219.5—HPCS, RCIC TRIP
165
(2) 129.8—Initiate RCIC, HPCS, start HPCS (Div 3) Diesel, trip Recirc. Pumps, close Primary System Isolation Valves except RHR shutdown Cooling Isolation Valves and MSIV's, provide signal to SBGTS.
(1) 16.5—Initiate RHR (LPCI Mode) and LPCS, Start Div I and Div II Diesels, Contribute to ADS, and close MSIV's
5

Narrow Range

230
(7) 205.1—High Level Alarm
(6) 197.1—Low Level Alarm
(3) 177.7—Reactor Scram, contribute to ADS, trip recirc pumps to LFMGS, close RHR shutdown cooling isolation valves.
165

REACTOR VESSEL WATER LEVEL
Bases Figure B 3/4 3-1

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager, Perry Plant ~~Operations~~ Nuclear Power Department (PNPPD), shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

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

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization 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 USAR and updated in accordance with 10 CFR 50.71(e).

b. The General Manager, Perry Plant ~~Operations~~ Nuclear Power Department (PNPPD) ~~(PPSD)~~, shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President - Nuclear ~~Group~~ 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. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

ADMINISTRATIVE CONTROLS

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of unit design and operating experience information, including units of similar design, which may indicate areas for improving unit safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving unit safety to the Director, Nuclear Engineering Department (PNED) Perry

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers or technically oriented individuals located onsite. Each shall have either (1) a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field, or (2) equivalent work experience as described in Section 4.1 of ANSI/ANS 3.1, December 1981.

RESPONSIBILITIES

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

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to the Director, Nuclear Engineering Department. Perry

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Plant Health Physicist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.

*Not responsible for sign-off function.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Perry Training Section Manager, and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 PLANT OPERATIONS REVIEW COMMITTEE (PORC)

FUNCTION

6.5.1.1 The PORC shall function to advise the General Manager, Perry Plant (PNPPD), Nuclear Power ~~Operations~~ Department ~~(PNPPD) and the Director, Perry Plant Technical Department (PPTD),~~ on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The PORC shall be composed of the:

Chairman:	Director, Perry Nuclear Engineering Department
Vice-Chairman/Member:	Manager, Perry Operations Section
Member:	Manager, ~~Technical Section~~ Systems Engineering Section
Member:	Manager, Perry Maintenance Section
Member:	Reactor Engineer
Member:	Manager, Radiation Protection Section
Member:	Plant Health Physicist
Member:	Manager, Instrumentation and Control Section
Member:	Manager, Licensing and Compliance Section

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PORC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PORC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

ADMINISTRATIVE CONTROLS

QUORUM

6.5.1.5 The quorum of the PORC necessary for the performance of the PORC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of all Administrative Procedures;

b. Review of the safety evaluations for (1) proposed procedures/instructions, (2) changes to procedures/instructions, equipment, systems or facilities, and (3) tests or experiments performed under the provisions of 10 CFR 50.59 to verify that such actions do not constitute an unreviewed safety question;

c. Review of proposed procedures/instructions and changes to procedures/instructions, equipment, systems or facilities which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Review of proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

e. Review of proposed changes to Technical Specifications or the Operating License;

f. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear ~~Group~~ and to the Nuclear Safety Review Committee;

g. Review of all REPORTABLE EVENTS;

h. Review of the plant Security Plan and Security Contingency Instructions;

i. Review of the Emergency Plan and implementing instructions;

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

k. Review of any accidental, unplanned or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the General Manager, Perry Plant ~~Operations~~ Department ~~(PPOD) and the Director, Perry Plant Technical Department (PPTD),~~ (Nuclear Power) (ANPPD), the Nuclear Safety Review Committee and the Vice President - Nuclear Group;

l. Review of Unit operations to detect potential hazards to nuclear safety;

m. Investigations or analysis of special subjects as requested by the Chairman of the Nuclear Safety Review Committee; and

n. Review of the Fire Protection Program and implementing procedures.

ADMINISTRATIVE CONTROLS

RESPONSIBILITIES (Continued)

6.5.1.7 The PORC shall:

a. Recommend in writing to the General Manager, ~~PNPPD Director, PNPP~~ PNPPD, approval or disapproval of items considered under Specifications 6.5.1.6a. through e., h., i., j., and k., above prior to their implementation;

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6b. through e., above, constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Vice President - Nuclear ~~Group~~ and the Nuclear Safety Review Committee of disagreement between the PORC and the General Manager, Perry ~~Plant Operations~~ Nuclear Power Department; however, the General Manager, Perry ~~Plant Operations~~ Nuclear Power Department, shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The PORC shall maintain written minutes of each PORC meeting that, at a minimum, document the results of all PORC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Vice President - Nuclear ~~Group~~ and the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The NSRC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,
b. Nuclear engineering,
c. Chemistry and radiochemistry,
d. Metallurgy,
e. Instrumentation and control,
f. Radiological safety,
g. Mechanical and electrical engineering,
h. Quality assurance practices and administrative controls, and
i. Nondestructive testing.

The NSRC shall report to and advise the Vice President - Nuclear ~~Group~~ on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

ADMINISTRATIVE CONTROLS

COMPOSITION

6.5.2.2 The membership of the NSRC shall be composed of at least eight personnel appointed by the Vice President - Nuclear Group to provide collective experience and competency in the following areas:

Nuclear power plant operations
Nuclear engineering
Chemistry and radiochemistry
Metallurgy
Nondestructive testing
Instrumentation and control
Radiological safety
Mechanical and electrical engineering
Administrative controls and quality assurance practices

The Chairman, appointed by the Vice President - Nuclear Group, shall have 10 years of power plant experience, of which 3 years shall be nuclear power plant experience. The NSRC members shall hold a bachelors' degree in an engineering or physical science field, or equivalent experience, and a minimum of 5 years of technical experience of which a minimum of 3 years shall be in one or more of the disciplines in Specification 6.5.2.1. Competent alternates may be designated in advance and consultants may be used for in-depth expertise if desired by the committee.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the NSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter.

QUORUM

6.5.2.6 The quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four but not less than one-half of the NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

ADMINISTRATIVE CONTROLS

AUDITS (Continued)

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

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified corporate licensee fire protection engineer(s) or an outside independent fire protection consultant. An outside independent fire protection consultant shall be utilized at least every third year;

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

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

k. Any other area of unit operation considered appropriate by the NSRC or the Vice President - Nuclear Group.

RECORDS

6.5.2.9 Records of NSRC activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved, and forwarded to the Vice President - Nuclear Group within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Vice President - Nuclear Group within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Vice President - Nuclear Group and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

ADMINISTRATIVE CONTROLS

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures/instructions required by Specification 6.8 and other procedures/instructions which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure/instruction or procedure/instruction change shall be reviewed by a qualified individual(s) other than the individual(s) which prepared the procedure/instruction or procedure/instruction change, but who may be from the same section as the individual(s) which prepared the procedure/instruction or procedure/instruction change. Instructions shall be approved by appropriate management personnel as designated in writing by PORC, and approved by the appropriate managers, ~~Perry Plant Departments. Both~~ the General Manager, ~~PPOD, and the Director, PPTD~~, PNPPD, shall approve Administrative Procedures. Section

b. Proposed modifications to plant structures, systems and components that affect nuclear safety shall be reviewed by individuals designated by the Director, Nuclear Engineering Department. Each such modification shall be reviewed by a qualified individual(s) other than the individual(s) which designed the modification, but who may be from the same section as the individual(s) which designed the modifications. Proposed modifications to plant structures, systems and components that affect nuclear safety shall be reviewed by PORC and approved prior to implementation by ~~both~~ the General Manager, ~~PPOD/Director, PPTD~~ PNPPD. Perry

c. Proposed tests and experiments which affect plant nuclear safety shall be prepared, reviewed, and approved. Each such test or experiment shall be reviewed by a qualified individual(s) other than the individual(s) which prepared the proposed test or experiment. Proposed tests and experiments shall be approved before implementation by ~~both~~ the General Manager, ~~PPOD/Director PPTD~~ PNPPD.

d. Sections responsible for reviews, including cross-disciplinary reviews, performed in accordance with Specifications 6.5.3.1a. and 6.5.3.1c., shall be designated in writing by PORC and approved by the General Manager, ~~PPOD, or the Director PPTD as appropriate~~. The individual(s) performing the review shall meet or exceed the qualification requirements of appropriate section(s) of ANSI N18.1-1971; PNPPD.

e. Each review shall include a determination pursuant to 10 CFR 50.59 of whether or not the potential for an unreviewed safety question exists. If such a potential does exist, a safety evaluation per 10 CFR 50.59 to determine whether or not an unreviewed safety question is involved shall be performed. Pursuant to 10 CFR 50.59, NRC approval of items involving unreviewed safety questions shall be obtained prior to implementation; and

ADMINISTRATIVE CONTROLS

ACTIVITIES (Continued)

f. The Plant Security Plan and Emergency Plan, and implementing instructions, shall be reviewed at least once per 12 months. ~~Recommended changes to the implementing instructions shall be approved by the Director, Nuclear Support Department, or Director, Perry Plant Technical Department, as appropriate.~~ Recommended changes to the Plans and implementing instructions shall be reviewed pursuant to the requirements of Specification 6.5.1.6 and approved by the General Manager, Perry Plant ~~Operations~~ Nuclear Power Department ~~and either the Director, Nuclear Support Department, or the Director, Perry Plant Technical Department, as appropriate~~. NRC approval shall be obtained as appropriate.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of the review submitted to the NSRC and the Vice President - Nuclear ~~Group~~.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear ~~Group~~ and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC, and the Vice President - Nuclear ~~Group~~ within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS

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

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

ADMINISTRATIVE CONTROLS

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS (Continued)

b. The applicable procedures required to implement the requirements of NUREG-0737 and supplements thereto.

c. Security Plan implementation.

d. Emergency Plan implementation.

e. PROCESS CONTROL PROGRAM implementation.

f. OFFSITE DOSE CALCULATION MANUAL implementation.

g. Radiological Environmental Monitoring Program implementation.

h. Fire Protection Program implementation.

6.8.2 Each administrative procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the PORC and shall be approved by the General Manager, PPOD, ~~and the Director, PNTD,~~ prior to implementation. All procedures/instructions shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes. Temporary changes to procedures/instructions which do not change the intent of the approved procedures/instructions shall be approved for implementation by two members of the plant management staff, at least one of whom holds a Senior Operator license. These temporary changes shall be documented. The temporary changes shall be approved by the original approval authority within 14 days. For changes to procedures/instructions which may involve a change in intent of the procedures/instructions, the original approval authority shall approve the change prior to implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the HPCS, RHR, RCIC, LPCS, feedwater leakage control system, the hydrogen analyzer portion of Combustible Gas Control, and post-accident sampling systems. The program shall include the following:

1. Preventive maintenance and periodic visual inspection requirements, and

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