UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. NPF-58

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated September 13, 1990, supplemented October 16, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 42 are hereby incorporated into this license. The Cleveland Electric Illuminating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall

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Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: March 20, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 42

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.

Remove	Insert
Index i	Index i
Index xxv	Index xxv
1-3	1-3
3/4 3-15	3/4 3-15
3/4 3-82	3/4 3-82
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B 3/4 0-1	B 3/4 0-1
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SECTION

SECTION

DEFINITIONS (Continued) PAGE

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.

FREQUENCY NOTATION

1.17 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

FUEL HANDLING BUILDING INTEGRITY

1.18 FUEL HANDLING BUILDING (FHB) INTEGRITY shall exist when:

a. The doors in each access to the 620 foot elevation of the FHB are closed, except for normal entry and exit.

b. The FHB railroad track door is closed.

c. The fuel handling area floor hatches are in place.

d. The FHB ventilation system is in compliance with Specification 3.7.7.1.

e. The shield blocks are installed adjacent to the Shield Building.

GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM

1.19 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgasses from the main condenser evacuation system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.20 IDENTIFIED LEAKAGE shall be:

a. Leakage into collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or

b. Leakage into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

ISOLATION SYSTEM RESPONSE TIME

1.21 The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation actuation setpoint at the channel sensor until the isolation valves travel to their required positions. 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.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

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.

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.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
ACTION

NOTES (Continued)

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the standby subsystem of the annulus exhaust gas treatment system.

(c) Also actuates the control room emergency filtration system in the recirculation mode of operation.

(d) Also trips and isolates the mechanical vacuum pumps.

(e) Closes only RWCU system isolation valve(s) 1G33-F004 (SLCS Pump A) and 1G33-F001 (SLCS Pump B).

(f) Manual initiation isolates 1E51-F064 and 1E51-F031 only and only following manual or automatic initiation of the RCIC system.

(g) Containment and Drywell Purge System inboard and outboard isolation valves each use a separate two out of two isolation logic.

(h) Requires RCIC system steam supply pressure - low coincident with drywell pressure high to isolate valve 1E51-F077.

(i) For this signal, one trip system has two channels which close valves 1E51-F063 and 1E51-F076 while the other trip system has two channels which close valve 1E51-F064.

(j) Isolates both RHR and RCIC.

(k) There is only one (1) RCIC manual initiation channel for valve group 9.

SOURCE RANGE MONITORS

LIMITING CONDITION FOR OPERATION

3.3.7.6 At least the following source range monitor channels shall be OPERABLE:

a. In OPERATIONAL CONDITION 2*, three.

b. In OPERATIONAL CONDITION 3 and 4, two.

APPLICABILITY: OPERATIONAL CONDITIONS 2*, 3 and 4.

ACTION:

a. In OPERATIONAL CONDITION 2* with one of the above required source range monitor channels inoperable, restore at least 3 source range monitor channels to OPERABLE status within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. In OPERATIONAL CONDITION 3 or 4 with one or more of the above required source range monitor channels inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

SURVEILLANCE REQUIREMENTS

4.3.7.6 Each of the above required source range monitor channels shall be demonstrated OPERABLE by:

a. Performance of a:

   1. CHANNEL CHECK at least once per:

      a) 12 hours in CONDITION 2*, and

      b) 24 hours in CONDITION 3 or 4.

   2. CHANNEL CALIBRATION** at least once per 18 months.

b. Performance of a CHANNEL FUNCTIONAL TEST:

   1. Within 24 hours prior to moving the reactor mode switch from the Shutdown position, if not performed within the previous 7 days, and

   2. At least once per 31 days.

c. Verifying, prior to withdrawal of control rods, that the SRM count rate is at least 0.7 cps*** with the detector fully inserted.

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*With IRM's on range 2 or below.

**Neutron detectors may be excluded from CHANNEL CALIBRATION.

***Provided the signal-to-noise ratio is $\geq$ 2.

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 8.7 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.

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 Specification 3.0.3 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 above applicable LPRM calibration and monitoring functions.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

d. 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.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.9 The plant systems actuation instrumentation channels shown in Table 3.3.9-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.9-2.

APPLICABILITY: As shown in Table 3.3.9-1.

ACTION:

a. With a plant system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.9-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. For the containment spray system:

1. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour or declare the associated system inoperable.

2. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, declare the associated system inoperable.

c. For the feedwater system/main turbine trip system:

1. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

2. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Declare the recirculation loop with the lower flow not in operation and take the ACTION required by Specification 3.4.1.1.

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

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

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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.3. and 4.6.1.8.4.

j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, and 4.6.1.2.d.

(Next page is 3/4 6-6.)

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.

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.

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.

PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Primary containment average air temperature shall not exceed 90°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the primary containment average air temperature greater than 90°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The primary containment average air temperature shall be the arithmetical average* of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	720'-6"	280°
b.	720'-6"	100°
c.	689'-4"	40°
d.	689'-4"	210°
e.	647'-0"	54°
f.	645'-6"	251°
g.	613'-0"	69°
h.	613'-0"	251°

*At least one reading from each elevation for an arithmetical average. However, all available instruments should be used in calculating the arithmetical average.

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 1000 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

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*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.

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 73,700 gallons of fuel for Div 1 and Div 2 and 36,100 gallons of fuel for Div 3, and

3. A separate fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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 once per 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 72 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.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit of the above required A.C. sources and diesel generator Div 1 or Div 2 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 once per 8 hours thereafter. If a diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators separately for each diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours* for each diesel generator which has not been successfully tested within the past 24 hours. Restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators Div 1 and Div 2 to OPERABLE status with 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With diesel generator Div 3 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the offsite A.C. 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 separately by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours*. Restore diesel generator Div 3 to OPERABLE status within 72 hours or declare the HPCS system and the C ESW pump inoperable and take the ACTION required by Specifications 3.5.1 and 3.7.1.1.

e. With diesel generator Div 1 or Div 2 of the above required A.C. electrical power sources inoperable, in addition to ACTION b or c, as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 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.

7. Verifying the pressure in all air start receivers for each diesel generator to be greater than or equal to 210 psig.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.

c. At least once per 92 days by checking for and removing accumulated water from the fuel oil storage tanks.

d. By sampling new fuel oil in accordance with ASTM D4057-88 prior to the addition to the storage tank and:

1. By verifying prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate; or an absolute specific gravity at 60/60°F, of greater than or equal to 0.83 but less than or equal to 0.89; or an API gravity at 60°F of greater than or equal to 26 degrees but less than or equal to 39 degrees, when tested in accordance with ASTM D1298-85,

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, when testing in accordance with the tests specified in ASTM D975-89, if gravity was not determined by comparison with the supplier's certification,

c) A flash point equal to or greater than 125°F, when tested in accordance with the tests specified in ASTM D975-89,

d) No visible free water or particulate contamination when tested in accordance with ASTM D4176-86.

2. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-89 are met when tested in accordance with the tests specified in ASTM D975-89.

e. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-88, and verifying that total particulate contamination is less than 10 mg/liter when tested in accordance with ASTM D2276-88.

f. At least once per 18 months*, ** during shutdown, by:

1. Subjecting the diesel to an inspection in accordance with instructions prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the diesel generator capability to reject a load of greater than or equal to 1400 kw (LPCS pump) for diesel generator Div 1, greater than or equal to 729 kw (RHR B pump or RHR C pump)

*For any start of a diesel, the diesel must be loaded in accordance with the manufacturer's recommendations.

**Except 4.8.1.1.2.f.1 to be performed every refueling outage, for the Div 1 and Div 2 diesel generators.

for diesel generator Div 2, and greater than or equal to 2400 kw (HPCS pump) for diesel generator Div 3 while maintaining speed less than nominal speed plus 75% of the difference between nominal speed and the overspeed trip setpoint or 15% above nominal, whichever is less.

3. Verifying the diesel generator capability to reject a load of 5800 kw for diesel generators Div 1 and Div 2 and 2600 kw for diesel generator Div 3 without tripping. The generator voltage shall not exceed 4784 volts for Div 1 and Div 2 and 5000 volts for Div 3 during and following the load rejection.

4. Simulating a loss of offsite power by itself, and:

   a) For divisions 1 and 2:

      1) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected loads through the load sequence (individual load timers) and operates for greater than or equal to 5 minutes while its generator is so loaded. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

   b) For division 3:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency bus with the permanently connected loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is so loaded. After energization, the steady

*All diesel generator starts for the purpose of this Surveillance Requirement may be preceded by an engine prelube period. The diesel generator start (10 sec)/load (60 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal for Div 1 and Div 2 and within 13 seconds after the auto-start signal for Div 3; the steady state generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal, and:

   a) For divisions 1 and 2:

      1) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

      2) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency loads and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

   b) For division 3:

      1) Verifying de-energization of the emergency bus.

      2) Verifying the diesel generator starts* on the auto-start signal, energizes the emergency bus with its loads and the auto-connected emergency loads within 13 seconds and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state

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*All diesel generator starts for the purpose of this Surveillance Requirement may be preceded by an engine prelube period. The diesel generator start (10 sec)/load (60 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other engine starts for the purpose of this surveillance testing may be preceded by other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

7. Verifying that all automatic diesel generator trips are automatically bypassed with an ECCS actuation signal except:

    a) For divisions 1 and 2, engine overspeed and generator differential current.

    b) For division 3, engine overspeed and generator differential current.

8. Verifying the diesel generator operates for at least 24 hours. During this test, the diesel generator shall be loaded to between 6800-7000 kw for the first two hours and between 5600-5800 kw** for the remaining 22 hours for diesel generator Div 1 and Div 2. The Div 3 diesel generator shall be loaded to greater than or equal to 2860 kw for the first two hours of this test and 2600 kw for the remaining 22 hours of this test. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal for Div 1 and Div 2 and within 13 seconds after the start signal for Div 3; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.f.4.a.2 and b.2* or perform Surveillance Requirement 4.8.1.1.2.f.6.a.2 and b.2.*

9. Verifying that the auto-connected loads to each diesel generator do not exceed 7000 kw for diesel generator Div 1 and Div 2 and 2860 kw for diesel generator Div 3.

10. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

11. Verifying that with the diesel generator operating in a test mode and connected to its bus, a simulated ECCS actuation signal overrides the test mode by (1) returning the diesel generator to

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*If Surveillance Requirements 4.8.1.1.2.f.4.a.2 and b.2 or 4.8.1.1.2.f.6.a.2 and b.2 are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator Div 1 or Div 2 may be operated at 5600-5800 kw or diesel generator Div 3 may be operated at 2600 kw for one hour or until operating temperatures have stabilized.

**This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test; the loads, however, shall not be less than 5600 kw nor greater than 7000 kw.

to standby operation, and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that each fuel transfer pump transfers fuel from the fuel storage tank to the day tank of each diesel.

13. Verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within $\pm$ 10% of its design interval for diesel generators Div 1 and Div 2.

14. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

    a. For diesel generators Div 1 and Div 2:

        1) Control room switch in pull-to-lock (with local/remote switch in remote).
        2) Local/remote switch in local
        3) Barring device engaged
        4) Inop/Normal switch in inop

    b. For diesel generator Div 3:

        1) Emergency run/stop switch in stop
        2) Maintenance/auto/test switch in maintenance
        3) Control room switch in pull-to-lock position

g. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that all three diesel generators accelerate to at least 441 rpm for diesel generators Div 1 and Div 2 and 882 rpm for diesel generator Div 3 in less than or equal to 10 seconds.

h. At least once per 10 years by:

    1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

    2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section 11 Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests of any diesel generator is greater than or equal to seven, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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
$\leq$ 1	$\leq$ 4	Once per 31 days
$\geq$ 2	$\geq$ 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.

**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).

A. C. SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator Div 1 or Div 2, and diesel generator Div 3 when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. A day 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 fuel storage system containing a minimum of 73,700 gallons of fuel for Div 1 and Div 2 and 36,100 gallons of fuel for Div 3.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the offsite circuits and/or diesel generators Div 1 or Div 2 of the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and Fuel Handling Building, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet 10 inches above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator Div 3 of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator Div 3 to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

c. With the fuel oil contained in the storage tank not meeting the properties specified in TS 4.8.1.1.2.d.2 or 4.8.1.1.2.e, the fuel oil shall be brought back within the specified limits within 7 days or the associated diesel generator shall be declared inoperable.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 (except for the requirement of 4.8.1.1.2.a.5), and 4.8.1.1.3.

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* When handling irradiated fuel in the Fuel Handling Building or primary containment.

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division 1, consisting of:
   1. 125 volt battery 1R42-S002 or 2R42-S002.
   2. 125 volt full capacity charger 1R42-S006 or 0R42-S007.

b. Division 2, consisting of:
   1. 125 volt battery 1R42-S003 or 2R42-S003.
   2. 125 volt full capacity charger 1R42-S008 or 0R42-S009.

c. Division 3, consisting of:
   1. 125 volt battery 1E22-S005 or 2E22-S005.
   2. 125 volt full capacity charger 1E22-S006 or 0R42-S011.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the Unit 1 and Unit 2 Division 1 batteries and/or both chargers of the above required Division 1 D.C. electrical power sources inoperable, restore an inoperable Division 1 battery and charger to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the Unit 1 and Unit 2 Division 2 batteries and/or both chargers of the above required Division 2 D.C. electrical power sources inoperable, restore an inoperable Division 2 battery and charger to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the Unit 1 and Unit 2 Division 3 batteries and/or both chargers of the above required Division 3 D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125 volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:
   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and
   2. Total battery terminal voltage is greater than or equal to 129 volts on float charge.

6.1 RESPONSIBILITY

6.1.1 The General Manager, Perry Nuclear Power Plant 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 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 Nuclear Power Plant Department (PNPPD), 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 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.

6.2.2 UNIT STAFF

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1;

b. At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Operator shall be in the control room;

c. A Health Physics Technician* shall be on site when fuel is in the reactor;

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

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physics technicians, auxiliary operators, and key maintenance personnel).

   The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).

*The Health Physics Technician may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

(Next page is 6-6)

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, Perry Nuclear Engineering Department (PNED).

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, Perry Nuclear Engineering Department.

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.

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*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 Nuclear Power Plant Department (PNPPD), 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, 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.

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 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 Nuclear Power Plant Department (PNPPD), the Nuclear Safety Review Committee and the Vice President - Nuclear;

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.

RESPONSIBILITIES (Continued)

6.5.1.7 The PORC shall:

a. Recommend in writing to the General Manager, 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 and the Nuclear Safety Review Committee of disagreement between the PORC and the General Manager, Perry Nuclear Power Plant Department; however the General Manager, Perry Nuclear Power Plant 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 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 on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

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.

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 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 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 and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

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 designed in writing by PORC, and approved by the appropriate Section managers. The General Manager, PNPPD, shall approve Administrative Procedures.

b. Proposed modifications to plant structures, systems and components that affect nuclear safety shall be reviewed by individuals designated by the Director, Perry 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 the General Manager, PNPPD.

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 the General Manager, 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, PNPPD. The individual(s) performing the review shall meet or exceed the qualification requirements or appropriate section(s) of ANSI N18.1-1971;

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

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 Plans and implementing instructions shall be reviewed pursuant to the requirements of Specification 6.5.1.6 and approved by the General Manager, Perry Nuclear Power Plant Department. 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.

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 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 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.

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, PNPPD, 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.

Specifications 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL CONDITIONS or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s)or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in an OPERATIONAL CONDITION or other specified condition in which the specification no longer applies.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered an OPERATIONAL CONDITION in which a new specification becomes applicable. In this case, the time limits of the ACTION requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirement within the specified time interval constitutes compliance with a specification, and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown CONDITION when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower CONDITIONS of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in COLD SHUTDOWN when a shutdown is required during POWER operation. If the plant is in a lower CONDITION of operation when a shutdown is required, the time limit for reaching the next lower CONDITION of operation applies. However, if a lower CONDITION of operation is reached in less time than allowed, the total allowable

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 OPERATIONAL CONDITIONS 3 and 4 to provide monitoring capability which provides diversity of protection to the mode switch interlocks.

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.

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 be 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.

MONITORING INSTRUMENTATION (Continued)

3/4.3.7.8 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.7.9 RADIOACTIVE LIQUID EFFLUENT MONITORING 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/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.7.10 RADIOACTIVE GASEOUS EFFLUENT MONITORING 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 setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the GASEOUS RADWASTE TREATMENT SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.8 TURBINE OVERSPEED PROTECTION SYSTEM

This specification is provided to ensure that the turbine overspeed protection system instrumentation and the turbine speed control valves are

OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION

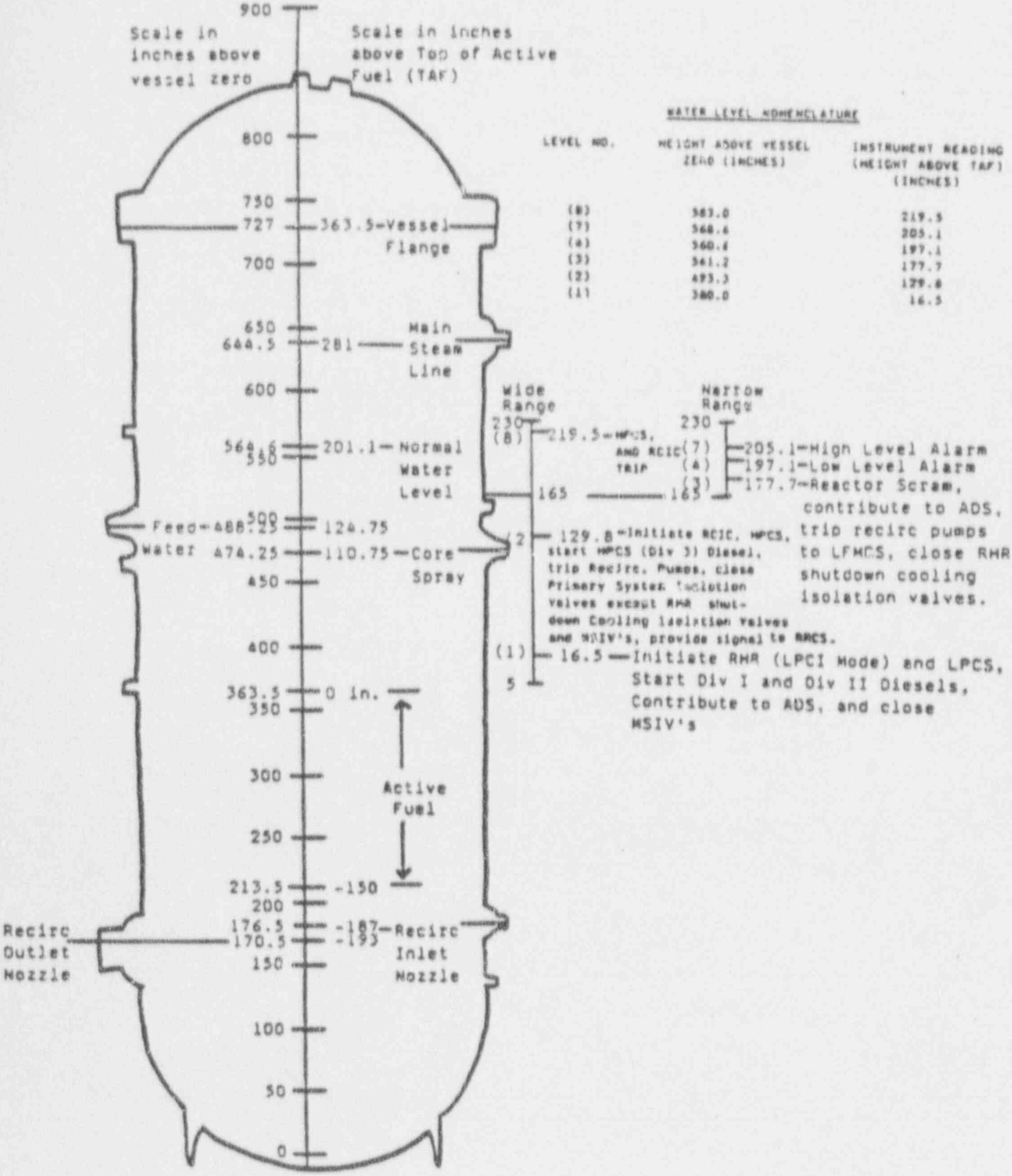
The plant systems actuation instrumentation is provided to initiate action of the containment spray system in the event of a LOCA with high containment

MONITORING INSTRUMENTATION (Continued)

3/4.3.9 PLANT SYSTEMS ACTUATION INSTRUMENTATION (Continued)

pressure and the feedwater system/main turbine trip system in the event of a failure of the feedwater controller under maximum demand. The LPCI mode of the RHR system is automatically initiated on a high drywell pressure signal and/or a low reactor water level, level 1, signal. The containment spray system will then actuate automatically following high drywell and high containment pressure signals. A 10-minute minimum and a 11.5-minute maximum time delay exists between initiation of LPCI and containment spray actuation. The suppression pool makeup system is automatically initiated on a low suppression pool water level signal with a concurrent LOCA signal or following a specified time delay after receipt of a LOCA signal.

A high reactor water level, level 8, signal will actuate the feedwater system/main turbine trip system.



WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	INSTRUMENT READING (HEIGHT ABOVE TAF) (INCHES)
(8)	583.0	219.5
(7)	568.6	205.1
(4)	560.6	197.1
(3)	541.2	177.7
(2)	493.3	129.8
(1)	380.0	16.5

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

SNUBBERS (Continued)

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

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.

3/4.7.6 MAIN TURBINE BYPASS SYSTEM

The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the feedwater controller failure analysis in FSAR Chapter 15.

3/4.7.7 FUEL HANDLING BUILDING

FUEL HANDLING BUILDING INTEGRITY ensures that the release of radioactive materials from the Fuel Handling Building following a fuel handling accident will be consistent with the accident analyses. The Fuel Handling Building Ventilation Exhaust System ensures that no significant fraction of the radioactive release from a postulated fuel handling accident could escape untreated.