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

YANKEE ATOMIC POWER COMPANY

YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-29

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 83
License No. DPR-3

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

   A. The application for amendment by Yankee Atomic Power Company (the licensee) dated May 26, 1981, as revised January 23, 1984 and February 26, 1985, 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. DPR-3 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 83, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 1, 1985.

ATTACHMENT TO LICENSE AMENDMENT NO. 83

FACILITY OPERATING LICENSE NO. DPR-3

DOCKET NO. 50-29

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and marginal lines indicating the area of change.

REMOVE	INSERT
2-1	2-1
2-3	2-3
2-5	2-5
B2-1	B2-1
B2-2	B2-2*
B2-3	B2-3
3/4 1-2	3/4 1-2
3/4 1-3	3/4 1-3*
3/4 1-4	3/4 1-4
3/4 2-1	3/4 2-1
3/4 3-3, 3/4 3-4	3/4 3-3, 3/4 3-4
3/4 3-12A	3/4 3-12A
3/4 3-13	3/4 3-13
3/4 3-28, 3/4 3-29	3/4 3-28, 3/4 3-29
3/4 3-34, 3/4 3-35	3/4 3-34, 3/4 3-35

* Overleaf page provided to maintain document completeness. No changes contained on this page.

3/4 4-2c	3/4 4-2c
3/4 5-7, 3/4 5-8	3/4 5-7, 3/4 5-8
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2*
3/4 7-3	3/4 7-3
3/4 7-4	3/4 7-4*
3/4 7-29a, b	3/4 7-29a, b, c, d
3/4 8-1	3/4 8-1
3/4 8-2	3/4 8-2
3/4 8-3, 3/4 8-4	3/4 8-3, 3/4 8-4
3/4 8-5	3/4 8-5
3/4 11-8, 3/4 11-9	3/4 11-8, 3/4 11-9
B3/4 7-1	B3/4 7-1
B3/4 7-4	B3/4 7-4
6-1	6-1*
6-2, 6-3	6-2, 6-3
6-6	6-6
6-11, 6-12, 6-13	6-11, 6-12, 6-13
6-15	6-15
6-18	6-18
6-24	6-24

* Overleaf page provided to maintain document completeness. No changes on this page.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, Main Coolant System pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1-1 for 4 loop operation.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop cold leg temperature and THERMAL POWER has exceeded (is above and to the right of) the appropriate Main Coolant System pressure line, be in HOT STANDBY within 1 hour.

MAIN COOLANT SYSTEM PRESSURE

2.1.2 The Main Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2

Whenever the Main Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Main Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Main Coolant System pressure has exceeded 2735 psig, reduce the Main Coolant System pressure to within its limit within 5 minutes.

BLANK

(INTENTIONALLY)

TABLE 2.2-1

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT
1. Manual Reactor Trip	Not Applicable
2. Power Range, Neutron Flux	Low Setpoint - $\leq$ 35% of RATED THERMAL POWER High Setpoint - $\leq$ 108% of RATED THERMAL POWER with 4 main coolant pumps operating
3. Intermediate Power Range, Neutron Flux	High Setpoint - $\leq$ 108% of RATED THERMAL POWER with 4 main coolant pumps operating
4. Intermediate Range, High Startup Rate	$\leq$ 5.2 decades/minutes
5. Source Range, Neutron Flux	Not Applicable
6. Low Main Coolant Flow (steam generator ΔP)	$\geq$ 80% of Design Flow
7. Low Main Coolant Flow (main coolant pump current)	$\geq$ 240 Amperes, $\leq$ 960 Amperes; with a response time of $\leq$500 msec.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the main coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and main coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Main Coolant System pressure and cold leg temperature for which the minimum DNBR is no less than 1.30, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. Because of flow instability, DNB may occur prematurely should the core exit quality become too great. The limiting core exit quality for preventing flow instability is taken conservatively at 0.08.

The limiting hot channel factors used in determining the thermal limit curves are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion.

SAFETY LIMITS

BASES

The curves are based on the following nuclear hot channel factors: F_q^N of 2.76; $F_{\Delta H}^N$ of 1.80; and a reference cosine with a peak of 1.44 for axial power shape.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion.

2.1.2 MAIN COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Main Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the main coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer and pumps are designed to Section VIII of the ASME Boiler and Pressure Vessel Code for Nuclear Power Plant, including all addenda through 1956, which permits a maximum transient pressure of 110%, 2735 psig, of design pressure. Pressure relief devices must be provided that will prevent pressure from exceeding 110 percent of the design pressure. The Main Coolant System piping and valves are designed to ANSI (formerly ASA) Standards, Power Piping Code, Section B31.1, 1955 Edition, and B16.5, 1957 Edition, respectively, which allows the design to be based on normal operating pressure and temperature and also allows exceeding the design conditions for periods of time. The stress level can be increased 15 percent above the Code allowable design value for not more than 10 percent of the design life and up to 20 percent above the allowable for up to 1 percent of the design life. Since normal plant operating pressure is 2000 psig, there is no conflict with either design condition. The setting of the Main Coolant System safety valves could allow pressure to increase to 2560 psig during a transient. The amount of time this condition is expected to exist is well within the allowances of B31.1. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Main Coolant System was hydrotested at 3435 psig, 138% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTIVE SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint limits specified in Table 2.2-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and Main Coolant System are prevented from exceeding their safety limits.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Power Range and Intermediate Power Range, Nuetron Flux

The Power Range and Intermediate Power Range Neutron Flux channel high setpoint provides reactor core protection against reactivity excursions which are too rapid to be protected by pressurizer water level protective circuitry. The Power Range low setpoint provides additional protection in the power range for a power excursion beginning from low power. The trip associated with the low setpoint may be manually bypassed above 15 MWe and is manually reinstated at a power level below 15 MWe. The low setpoint trip is not assumed in the accident analysis.

The prescribed setpoint, with allowances for errors, is consistent with the trip point used in the accident analysis.

Intermediate Range, Neutron Flux, High Startup Rate

The Intermediate Range High Startup Rate trip provides protection to limit the rate of power increase during low power conditions in the event of an uncontrolled rod withdrawal.

SURVEILLANCE REQUIREMENTS (Continued)

e. Factors to consider:

1. Main Coolant System boron concentration,

2. Control rod position,

3. Main Coolant System average temperature,

4. Fuel burnup based on gross thermal energy generation,

5. Xenon concentration, and

6. Samarium concentration.

4.1.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 0.8% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN (with all control rods inserted) shall be $\geq$ 5.0% $\Delta k/k$.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the SHUTDOWN MARGIN (with all control rods inserted) < 5.0% $\Delta k/k$, immediately continue boration at $\geq$ 26 gpm of $\geq$ 2200 ppm boron concentration or equivalent and establish and maintain CONTAINMENT INTEGRITY until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2.1 The SHUTDOWN MARGIN (with all control rods inserted) shall be determined to be $\geq$ 5.0% $\Delta k/k$:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. At least once per 24 hours by consideration of the following factors:

   1. Main Coolant System boron concentration,

   2. Control rod position,

   3. Main Coolant System average temperature,

   4. Fuel burnup based on gross thermal energy generation,

   5. Xenon concentration, and

   6. Samarium concentration.

SURVEILLANCE REQUIREMENTS (Continued)

This page intentionally left blank

3/4.2 POWER DISTRIBUTION LIMITS

PEAK LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The peak linear heat generation rate (LHGR) shall not exceed the limits of Figure 3.2-1 during steady state operation.

APPLICABILITY: MODE 1.

ACTION:

With the peak LHGR exceeding the limits of Figure 3.2-1:

a. Within 15 minutes reduce THERMAL POWER to not more than that fraction of the THERMAL POWER allowable for the main coolant pump combination in operation, as expressed below:

$$\text{Fraction of THERMAL POWER} = \frac{\text{Limiting LHGR}}{\text{Peak Full Power LHGR}}$$

b. Within 4 hours reduce the Power Range and Intermediate Power Range Neutron Flux high trip setpoint to $\leq$ 108% of the fraction of THERMAL POWER allowable for the main coolant pump combination.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The peak LHGR shall be determined to be within the limits of Figure 3.2-1 using incore instrumentation to obtain a power distribution map:

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

b. At least once per 1,000 EFPH,

c. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Turbine Trip	1	1	1	1(3)(6)	8
12. Generator Trip	1	1	1	1(3)(7)	8
13. Reactor Trip Breaker	2	1	2	1, 2 and *	9
14. Automatic Trip Logic	2	1	2	1, 2 and *	9
15. Main Steam Isolation Trip Logic	2	1	2	1, 2(4)	8

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal.

** The provisions of Specification 3.0.4 are not applicable.

High voltage to detector is automatically de-energized above 5×10^{-9} Amperes on the Intermediate Range.

Or when other activities might increase reactivity.

(1) Power Range Neutron Flux Low Setpoint Trip may be manually bypassed at $\geq$ 15 MWe. Bypass shall be manually removed prior to decreasing below 15 MWe.

(2) Intermediate Range Neutron Flux High Startup Rate Trip is automatically bypassed $\geq$ 15 MWe. Bypass is automatically removed prior to decreasing below 15 MWe.

(3) Trip may be manually bypassed $\leq$ 15 MWe. Bypass is automatically removed prior to increasing above 15 MWe.

(4) Trip may be manually bypassed when the reactor is not critical.

(5) Startup rate alarm setpoint $\leq$ 1.1 decade/minute.

(6) Turbine shall be protected by at least the following protective trips: rotor excessive axial movement, low bearing oil pressure; low condenser vacuum; and overspeed.

(7) Generator shall be protected by at least the following protective trips: overcurrent; differential; and loss of field.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

TABLE 3.3-2 (Continued)

ENGINEERING SAFEGUARDS SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS AND SENSORS	CHANNELS AND SENSORS TO TRIP	MINIMUM CHANNELS AND SENSORS OPERABLE	APPLICABLE MODES	ACTION
2. CONTAINMENT ISOLATION (Continued)					
c. Actuation Channel B	1	1	1	1, 2, 3, 4, 5(1)	10
1) High Containment Pressure Sensor	1	1	1	1, 2, 3, 4, 5(1)	10
2) Safety Injection	(All Safety Injection Initiating Functions and Requirements)				
3. MAIN STEAM ISOLATION					
a. Low Steam Line Pressure	3/Steam Line	2/Steam Line	3/Steam Line	1, 2, 3(2)	6**
b. Automatic Trip Logic	2	1	2	1, 2, 3(2)	8
c. Manual Initiation	2	1	2	1, 2	8
d. High Containment Pressure Trip Containment Isolation	2	1	2	1, 2	8

TABLE 3.3-2 (Continued)

TABLE NOTATION

** The provisions of Specification 3.0.4 are not applicable.

(1) Trip function may be bypassed in this MODE with main coolant pressure <300 psig.

(2) Trip function may be bypassed in this MODE with main coolant pressure <1800 psig and main coolant temperature <490°F.

(3) Automatic initiation of Actuation Channel #1 may be bypassed in this MODE during functional test of the Main Coolant System Loop 1 pressure channel.

Automatic initiation of Actuation Channel #2 may be bypassed in this MODE during functional test of the Main Coolant System Loop 2 pressure channel.

ACTION STATEMENTS

ACTION 10 - With the number of OPERABLE channels or sensors one less than the total number of channels or sensors, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one safety injection channel high containment pressure sensor may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 6 - With the number of OPERABLE channels one less than the total number of channels, STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

1. The inoperable channel is placed in the tripped condition within 1 hour.
2. The minimum channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 8 - With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

TABLE 3.3-6

FIRE DETECTION INSTRUMENTS

INSTRUMENTATION LOCATION	MINIMUM INSTRUMENTS OPERABLE
1. Control Room	
Above Dropped Ceiling	9
Control Boards	
Main Control Board	3
SI Panels	1/Panel
General Area	9
2. Cable Spreading	
Cable Tray House	2
Manhole No. 3	1
3. Switchgear Room	20
Battery Room No. 1	1
Battery Room No. 2	1
4. Diesel Generators	
No. 1	1
No. 2	1
No. 3	1
5. Safety Injection Pumps and No. 3 Battery	5
6. Charging Pump Cubicles	
No. 1	1
No. 2	1
No. 3	1
7. 1 & 2 Charcoal Filters	1/Filter
8. Turbine Building	
Transformer Oil Cooler Area	2
Turbine Lube Oil Reservoir	2
9. Vapor Container	1
10. PICS Building	1
11. Non-Return Valve (NRV) Enclosure	2

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The accident monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-7, either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-7, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-5.

TABLE 3.3-8 (Continued)

ACTION STATEMENTS

Action 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases from the tank may continue provided that prior to initiating the release:

1. At least two independent samples of the tank's contents are analyzed in accordance with Specification 4.11.1.1.1;

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

otherwise, suspend release of radioactive effluents via this pathway.

Action 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 1E-07 microcuries/gram:

a. At least once per 8 hours when the specific activity of the secondary coolant is greater than 0.01 microcuries/gram I-131.

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microcuries/gram I-131.

Action 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per four hours during actual releases. (Pump curves may be used to estimate flow.)

TABLE 4.3-6

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation					
a. Liquid Radwaste Effluents Line	D	P	R(2)	Q(1)	At all times
b. Steam Generator Blowdown Tank Effluent Line	D	M	R(2)	Q(1)	At all times
2. Continuous Composite Samplers and Sample Flow Measurement Device					
a. Steam Generator Blowdown Tank Effluent Line	D	N.A.	R	Q	**
b. Secondary Coolant and Condensate Leakage	D	N.A.	R	Q	**
c. Turbine Building Sump	D	N.A.	R	Q	**
3. Flow Rate Measurement Devices					
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q	**
b. Circulating Water System Discharge*	D(3)	N.A.	N.A.	N.A.	**
c. Steam Generator Blowdown Tank Effluent	N.A.	N.A.	R(4)	N.A.	**

*Pump curves utilized for flow rate determination.
**During releases via this pathway.

MAIN COOLANT SYSTEM

LIMITING CONDITIONS FOR OPERATION (Continued)

c. With the reactor vessel and connecting pressurizer system isolated from the heat removal system by closing the loop isolation valve(s), leak testing may be performed provided that the coolant temperature in the reactor vessel does not increase at a rate exceeding 50° per hour, the maximum temperature increase during the test period does not exceed $100^{\circ}F$, and pressurizer pressure does not exceed 2485 psig.

SURVEILLANCE REQUIREMENTS

4.4.1.1.3.1 The required residual heat removal loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.1.3.2 The required main coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.1.3.3 Determine that the steam generator(s) associated with the main coolant loop(s) required to be in operation are capable of decay heat removal by verifying at least once per 24 hours that:

a. The Main Coolant System is closed and pressurized to $\geq$100 psi above saturation pressure.

b. The Main Coolant System loop cold and hot leg stop valves are fully open, with the bypass valve closed.

c. The steam generator water level is above the top of the tube bundle.

d. An inventory of over 85,000 gallons of primary grade feedwater is available.

e. A boiler feed pump is OPERABLE.

4.4.1.1.3.4 At least one coolant loop shall be verified to be in operation and circulating main coolant at least once per 12 hours.

4.4.1.1.3.5 Verify that the Shutdown Cooling System isolation valves are locked closed within one hour prior to increasing Main Coolant System pressure above 300 psig.

4.4.1.1.3.6 At least once per 18 months, during shutdown, demonstrate main coolant loop isolation valve operability by cycling each valve through at least one complete cycle of full travel from the control room.

SURVEILLANCE REQUIREMENTS (continued)

5. Verifying that each ECCS safety injection subsystem is aligned to receive electrical power from an OPERABLE emergency bus.

6. Verifying that each pair of ECCS recirculation subsystem redundant valves is aligned to receive electrical power from separate OPERABLE busses.

7. Verifying that each pair of ECCS long-term hot leg injection subsystem redundant valves is aligned to receive electrical power from separate OPERABLE busses.

8. Verifying that the charging header flow metering instrument is OPERABLE by observing charging flow rate at least once per 12 hours.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the containment sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1. For all accessible areas of the containment prior to establishing containment integrity, and

2. Of the areas affected within containment at the completion of each containment entry when containment integrity is established.

d. At least once per 18 months by visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

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

1. Cycling each power-operated valve in the flow path through at least one complete cycle of full travel.

2. Verifying that valve CS-MOV-532 actuates to its correct position on a safety injection signal.

3. Verifying that each of the following pumps start automatically upon receipt of a safety injection signal:

(a) High Pressure Safety Injection (HPSI) pump

(b) Low Pressure Safety Injection (LPSI) pump

4. Verifying that two low pressure safety injection pumps develop a combined flow $\geq$ 2180 gpm. Test every LPSI pump at least once per 36 months.

5. Verifying that each charging pump stops automatically upon receipt of a safety injection signal.

6. Verifying that the charging header flow metering instrument is OPERABLE by performing a CHANNEL CALIBRATION.

7. Verifying the proper positioning of the HPSI throttle valves SI-V-671, 672, 673, and 674 by performing an inspection to ensure that:

   (a) Each valve locking device is in place and securely welded to the valve handle and to the valve yoke.

   (b) The scribe mark on each valve body aligns with the scribe mark on the valve yoke.

8. Verifying the proper positioning of hot leg injection throttle valve SI-V-645 at least once per 36 months by flow testing.

f. At least every 36 months, and/or any time either test under 4.5.2.e.7 is failed, by developing a backpressure of 875 psig in the high pressure safety injection header with two HPSI pumps operating as follows:

1. Pressure to the suction of the HPSI pumps to be 170 $\pm$ 10 psi.

2. LPSI flow is isolated.

3. Injection flow is to one loop with the other loops isolated by closing the appropriate injection gate valves CS-MOV-536, CS-MOV-537, CS-MOV-538, and CS-MOV-539.

4. The flow to the injection loops shall not be less than 200 gpm.

5. The above test shall be repeated to include the operation of all HPSI pumps.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator of an unisolated main coolant loop shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 4 main coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable:

   1. Operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either:

      a) The inoperable valve(s) is restored to OPERABLE status, or

      b) Three Power Range Neutron Flux channels are OPERABLE** with:

         1) The Power Range coincidence selector switch in the single position,

         2) The trip setpoints reduced per:

            (a) Table 3.7-1 for 4 loop operation.

         3) One Intermediate Power Range Neutron Flux channel in the tripped condition.

   2. Oth rwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 4.7-1, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, and Addenda through Summer, 1975.

** One Power Range Neutron Flux channel may be made inoperable for up to 2 hours for required surveillance per Specification 4.3.1.1.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT
WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 4 LOOP OPERATION

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1	27

BLANK

(INTENTIONALLY)

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	LIFT SETTING ($\pm$ 3%)	ORIFICE SIZE*
a. SV-409 E, F, G or H	935 psig	K
b. SV-409 A, B, C or D	985 psig	K_2
c. SV-409 I, J, K or L	1035 psig	Q

*K = 1.838 square inches
K_2 = 2.545 square inches
Q = 11.05 square inches

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load the ability of the snubber to withstand load without displacement shall be verified.

e. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.n.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Table 3.7-4 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement, or reconditioning shall be indicated in the records.

TABLE 3.7-4

SAFETY-RELATED MECHANICAL SNUBBERS*

Snubber No.	System Snubber Installed On Location	Accessible or Inaccessible (A or I)	High Radiation Zone During Shutdown** (Yes or No)	Especially Difficult to Remove (Yes or No)
CRU-SNB-19A	Pressurizer Relief Valve, PR-SV-182	A	No	No
CRU-SNB-19B	Pressurizer Relief Valve, PR-SV-182	A	No	No
CRU-SNB-20A	Pressurizer Relief Valve, PR-SV-181	A	No	No
CRU-SNB-20B	Pressurizer Relief Valve, PR-SV-181	A	No	No
SG-SNB-1R	S/G No. 1, Right Side	I	Yes	Yes
SG-SNB-1L	S/G No. 1, Left Side	I	Yes	Yes
SG-SNB-2R	S/G No. 2, Right Side	I	Yes	Yes
SG-SNB-2L	S/G No. 2, Left Side	I	Yes	Yes
SG-SNB-3R	S/G No. 3, Right Side	I	Yes	Yes
SG-SNB-3L	S/G No. 3, Left Side	I	Yes	Yes
SG-SNB-4R	S/G No. 4, Right Side	I	Yes	Yes
SG-SNB-4L	S/G No. 4, Left Side	I	Yes	Yes
PRSH-SNB-1	Safety Injection in VC - Loop 3	A	Yes	No
PRSH-SNB-2	Safety Injection in VC - Loop 3	A	Yes	No

TABLE 3.7-4 (Continued)

Snubber No.	System Snubber Installed On Location	Accessible or Inaccessible (A or I)	High Radiation Zone During Shutdown** (Yes or No)	Especially Difficult to Remove (Yes or No)
PRSH-SNB-3	Safety Injection in VC - Loop 1	A	Yes	No
PRSH-SNB-4	Safety Injection in VC - Loop 1	A	Yes	No
PRSH-SNB-5	Safety Injection in VC - Loop 2	A	Yes	No
PRSH-SNB-6	Safety Injection in VC - Drain Box	A	Yes	No
PRSH-SNB-7	Safety Injection in VC - Loop 4	A	Yes	No
PRSH-SNB-8	Safety Injection in VC - Loop 4	A	Yes	No
PRSH-SNB-9	Safety Injection in VC - Drain Box	A	Yes	No
PRSH-SNB-10	Safety Injection in VC - Drain Box	A	Yes	No
HPSI-SNB-1	Safety Injection in VC - Drain Box	A	Yes	No
HPSI-SNB-2	Safety Injection in VC - Drain Box	A	Yes	No
HPSI-SNB-3	Safety Injection in VC - Drain Box	A	Yes	No
PRCH-SNB-1A	Shutdown Cooling in VC - to Loop 4	A	Yes	No

TABLE 3.7-4 (Continued)

Snubber No.	System Snubber Installed On Location	Accessible or Inaccessible (A or I)	High Radiation Zone During Shutdown** (Yes or No)	Especially Difficult to Remove (Yes or No)
PRCH-SNB-1B	Shutdown Cooling in VC - to Loop 4	A	Yes	No
PRCH-SNB-2A	Shutdown Cooling in VC - to Loop 4	A	Yes	No
PRCH-SNB-2B	Shutdown Cooling in VC - to Loop 4	A	Yes	No
PRCH-SNB-3A	Shutdown Cooling in VC - to Loop 4	A	Yes	No
PRCH-SNB-3B	Shutdown Cooling in VC - to Loop 4	A	Yes	No
PRCH-SNB-4A	Shutdown Cooling in VC - to Loop 4	A	Yes	No
PRCH-SNB-4B	Shutdown Cooling in VC - to Loop 4	A	Yes	No

* Snubbers may be added to safety-related systems without prior License Amendment to Table 3.7-4 provided that a revision to Table 3.7-4 is included with the next License Amendment request.

** Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4 is included with the next License Amendment request.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 AC SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following AC 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:

1. Each with separate day fuel tank containing a minimum volume of 210 gallons of fuel, equivalent to a 3/4 full tank, and

2. With a fuel storage system containing a minimum volume of 8000 gallons of fuel, equivalent to a tank level of 4'6.5".

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

ACTION:

a. With either an offsite circuit or diesel generator of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.5 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required AC electrical power sources inoperable, demonstrate the OPERABILITY of the remaining AC sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.5 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30

LIMITING CONDITION FOR OPERATION (Continued)

hours. Restore at least two offsite circuits and three diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of three diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.5 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With less than two of the above required diesel generators OPERABLE demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least two of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore three diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by manually transferring unit power supply from one independent circuit to the second independent circuit.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day fuel tank,

2. Verifying gravity flow from the storage system to the day tanks,

3. Verifying the diesel starts from ambient condition and th generator voltage reaches $\geq$432 volts within 14 sec nds,

4. Verifying the generator is synchronized, loaded to $\geq$200 kW, and operates for $\geq$2 hours, and

5. Verifying the diesel generator is aligned to provide standby power to the associated emergency buses.

b. At least once per 31 days by verifying the fuel level in the fuel storage tank,

c. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment,

d. At least once per 18 months during shutdown by:

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

2. Verifying the generator capability to reject a load of $\geq$275 amperes without tripping,

3. Simulating a loss of off-site power in conjunction with a safety injection test signal, and:

a) Verifying de-energization of the emergency buses.

SURVEILLANCE REQUIREMENTS (Continued)

b) Verifying the diesel starts from ambient condition on the auto-start signal, the diesel generator voltage reaches $\geq$432 volts within 14 seconds, energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$5 minutes while its generator is loaded with the emergency loads.

4. Verifying the diesel generator operates for $\geq$60 minutes while loaded to $\geq$400 kW.

5. Verifying that the high pressure safety injection pump breakers on each emergency bus delay 10 $\pm$ 3 seconds in closing on the bus.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following AC electrical power sources shall be OPERABLE:

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

b. One diesel generator with:

1. Day fuel tank containing a minimum volume of 210 gallons of fuel, equivalent to a 3/4 full tank, and

2. A fuel storage system containing a minimum volume of 4000 gallons of fuel, equivalent to a tank level of 2'4.5".

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required AC electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required AC electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required AC electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for Requirement 4.8.1.1.2.a.4.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	LLD uCi/ml(a)
A. Waste Gas Surge Drum (one only)	P Grab Sample	P	Principal Gamma Emitters[b]	1E-04
B. Containment PURGE	P Each PURGE Grab Sample	P Each PURGE	Principal Gamma Emitters[b]	1E-04
			H-3	1E-06
C. Plant Vent (1) (Primary Vent Stack)	M(c) Grab Sample	M(c)	Principal Gamma Emitters[b]	1E-04
			H-3	1E-06
	Continuous[e]	W(d) Radioiodine Canister	I-131	1E-12
	Continuous[e]	W(d) Particulate	Principal Gamma Emitters[b] I-131	1E-11
	Continuous[e]	M Composite Particulate Sample	Gross Alpha	1E-11
	Continuous[e]	Q Composite Particulate Sample	Strontium 89, 90	1E-11
	Continuous[e]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	$1x10^{-5}$

TABLE 4.11-2 (Continued)

TABLE NOTATION

NOTE 1 - The ventilation header channels air through the ventilation system to the plant vent stack. The following ventilation systems discharge directly into the ventilation header.

- o Auxiliary Building
- o Radwaste Building
- o Containment Purge
- o Fuel Storage Area
- o Condenser Air Ejector
- o Waste Gas Holdup System Discharge

The steam generator blowdown vent discharges directly into the plant vent stack.

a. The lower limit of detection (LLD) is defined in Table Notation a. of Table 4.12-1 of Specification 4.12.1.1.

b. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide.

c. Sampling and analysis shall also be performed following a THERMAL POWER change of greater than 15 percent of RATED THERMAL POWER within one hour. A grab sample for noble gas analysis shall be taken within 8 hours and analyzed within 24 hours of the THERMAL POWER change. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased more than a factor of 3.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing or after removal from samplers. Sampling shall also be performed at least once per 24 hours for at least 7 days following a THERMAL POWER change of greater than 15 percent of RATED THERMAL POWER within one hour. Samples collected for 24 hours will be analyzed within 48 hours of changing, and the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration of the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that the effluent activity has not increased more than a factor of 3.

e. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1035 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Code, 1956 Edition. The total relieving capacity for all valves on all of the steam lines is 3.1×10^6 lbs/hr which is 129 percent of the total secondary steam flow of 2.4×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per OPERABLE steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are derived on the following bases:

For 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (108)$$

Where:

SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = Maximum number of inoperable safety valves per steam Generator

PLANT SYSTEMS

BASES

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations are based on a steam generator initial RT_{NDT} plus $60^{o}F$ and are sufficient to prevent brittle fracture.

3/4.7.3 PRIMARY PUMP SEAL WATER SYSTEM

(Deleted)

3/4.7.4 SERVICE WATER SYSTEM

(Deleted)

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM EMERGENCY SHUTDOWN

The operability of the control room ventilation system emergency shutdown enhances the opportunity for the control room to remain habitable for Operations personnel during and following accident conditions.

3/4.7.6 SEALED SOURCE CONTAMINATION

The limitation 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 or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

6.0 ADMINISTRATIVE CONTROLS

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

6.1 RESPONSIBILITY

6.1.1 The Plant Superintendent shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 In all matters relating to the operation of the plant and to these Technical Specifications, the Plant Superintendent shall report to and be directly responsible to the Manager of Operations in the Yankee Atomic Electric Company.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor. Operating crew personnel trained in radiation protection procedures fill this requirement.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed senior reactor operator or senior reactor operator limited to fuel handling who has no other concurrent responsibilities during this operation.

f. A fire brigade of at least 5 members shall be maintained on-site at all times. The fire brigade shall not include the minimum shift crew necessary for safe shutdown of the plant, 2 licensed operators, or any personnel required for other essential functions during a fire emergency.

g. Administrative procedures shall be developed and implemented to limit the working hours of the unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

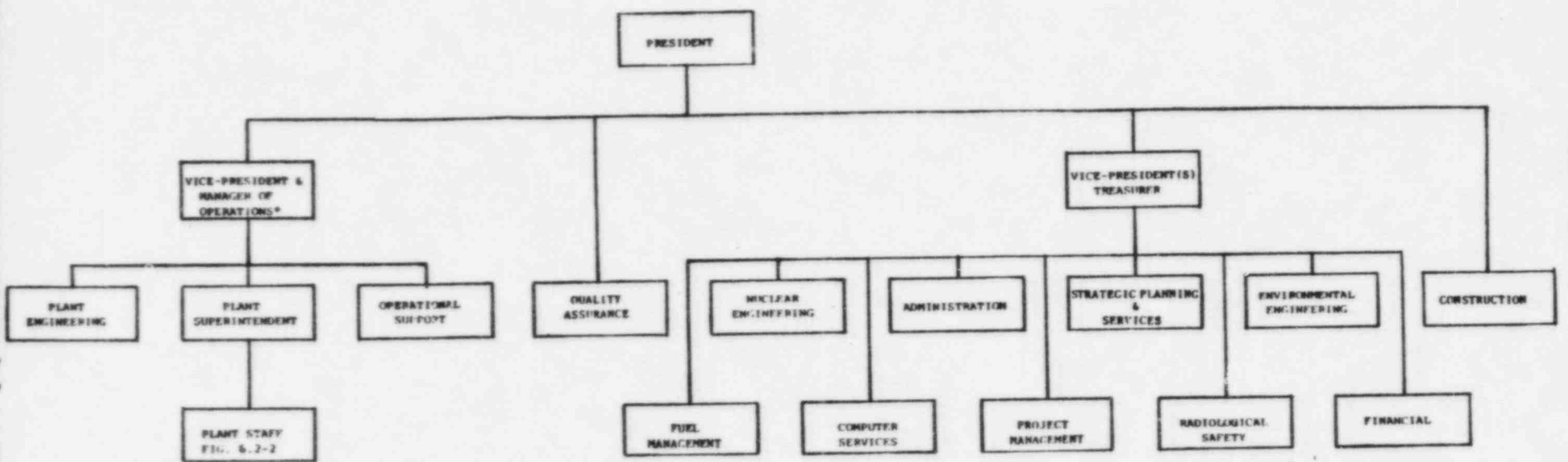
1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Superintendent or his delegated representative, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. ontrols shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Superintendent or his delegated representative to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.



* INCLUDES FIRE PROTECTION AND SECURITY RESPONSIBILITIES

Figure 6.2-1

OFFSITE ORGANIZATION

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff listed below shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents, and the Radiation Protection Manager who shall meet the minimum qualifications of Regulatory Guide 1.8, Revision 1.

a. Plant Superintendent

b. Assistant Plant Superintendent

c. Plant Chemistry Manager

d. Plant Operations Manager

e. Reactor Engineering Manager

f. Plant Maintenance Manager

g. Maintenance Supervisor

h. Instrument and Control Supervisor

i. Shift Supervisors

j. Radiation Protection Manager

k. Technical Director

l. Shift Technical Advisor

m. Technical Services Supervisor

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility NRC licenses staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of a member of the plant staff appointed to perform the duties of Fire Protection Coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975, except for Fire Brigade training sessions which shall be held at least quarterly.

b. Proposed changes to equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed test or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety, defined as Plant Information Reports.

g. Events requiring 24-hour written notification to the Commission.

h. Reports and meeting minutes of the Plant Operation Review Committee.

i. Perform special reviews and investigations and render reports thereon as requested by the Vice President of Operations.

AUDITS

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the NSAR Committee. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months, $\pm 25\%$.

b. The performance, training and qualification of those members of the facility staff who have a direct relationship to operation, maintenance or technical aspects of the plant, at least once per 12 months, $\pm$ 25%.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months, $\pm$ 25%.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months, $\pm$ 25%.

e. The Facility Emergency Plan and implementing procedures at least once per 12 months, $\pm$ 25%.

f. The Facility Security Plan and implementing procedures at least once per 12 months, ± 25%.

g. The Facility Fire Protection Program and implementing procedures at least once per 24 months, ± 25%.

h. The radiological environmental monitoring program and the results thereof at least once per 12 months, ± 25%.

i. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months, ± 25%.

j. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive waste at least once per 24 months, ± 25%.

k. The performance of activities required by the Quality Assurance Program to meet the provisions of Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975 at least once per 12 months, ± 25%.

l. Any other area of facility operation considered appropriate by the NSAR Committee or the Vice President.

AUTHORITY

6.5.2.10 The NSAR Committee shall report to and advise the Vice President on those areas of responsibility specified in Sections 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Minutes of each NSAR Committee meeting shall be prepared and forwarded to the Vice President and each member of the Committee for review within 20 working days following each meeting. The meeting minutes shall include, where applicable, reports of reviews encompassed by Section 6.5.2.8; and reports of audits encompassed by Section 6.5.2.9. The review of the minutes shall be completed within 60 days of the date of their distribution.

6.5.3 INDEPENDENT AUDIT AND REVIEW

6.5.3.1 An independent fire protection and loss prevention program inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

6.5.3.2 An inspection and audit of the fire protection and loss prevention program shall be performed by a qualified outside fire consultant at least once per 36 months.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the PORC and submitted to the NSAR Committee and the Manager of Operations.

6.7 SAFETY LIMIT VIOLATION

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

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Manager of Operations and to the NSAR Committee within 24 hours.

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

d. The Safety Limit Violation Report shall be submitted to the Commission, the NSAR Committee and the Manager of Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained that meet or exceed the requirements and recommendations of Sections 5.2-5.2.9 and 5.3 of ANSI N18.7-1976 and Appendix "A" of Regulatory Guide 1.33, Revision 2, except as provided in 6.8.2 and 6.8.3 below. The written procedures shall also cover the activities relating to:

a. FIRE PROTECTION PROGRAM implementation.
b. PROCESS CONTROL PROGRAM implementation.
c. OFFSITE DOSE CALCULATION MANUAL implementation.
d. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974 and Regulatory Guide 4.1, Revision 1, April 1975.

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

6.9.2 Annual Report. Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions[a]; e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the whole body dose received from external sources shall be assigned to specific major work functions.

b. Any other unit-unique reports required on an annual basis.

c. Documentation of all challenges to the pressurizer Power-Operated Relief Valves (PORVs) or safety valves.

6.9.3 Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report.

(a) This tabulation supplements the requirements of Section 20.407 of 10CFR Part 20.

Note: This item is intended to provide for reporting of potentially generic problems.

(10) Exceeding the limits in Specification 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and description of activities planned and/or taken to reduce the contents to within the specified limits.

(11) Failure of the pressurizer PORVs or safety valves.

b. Thirty Day Written Reports. The REPORTABLE OCCURRENCES discussed below shall be the subject of written reports to the Director of the appropriate Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide a complete explanation of the circumstances surrounding the event.

(1) Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

(2) Conditions leading to operation in a degraded mode permitted by a limiting condition for operation or plant shutdown required by a limiting condition for operation.

Note: Routine surveillance testing, instrument calibration, or preventative maintenance which require system configurations as described in Items b(1) and b(2) need not be reported except where test results themselves reveal a degraded mode as described above.

(3) Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

(4) Abnormal degradation of systems other than those specified in Item a(3) above designed to contain radioactive material resulting from the fission process.

n. Records of the service lives of all mechanical snubbers listed on Table 3.7-4, including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposures.

6.12 HIGH RADIATION AREA

6.12.1 Paragraph 20.203 "Caution signs, labels, signals, and controls". In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2), each high radiation area in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area, and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit.* An individual or group of individuals permitted to enter such areas shall be provided with one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area, and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A Health Physics qualified individual (i.e., qualified in radiation protection procedures), with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and who will perform radiation surveillance at the frequency specified in the RWP. The surveillance frequency will be established by the Plant Health Physicist.

* Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, providing they are following plant radiation protection procedures for entry into high radiation areas.