October 9, 1985

Docket No. 50-271

See Correction letter of 3/17/86

Mr. R. W. Capstick
Licensing Engineer
Vermont Yankee Nuclear Power Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Capstick:

The Commission has issued the enclosed Amendment No. 90 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications in response to your application March 27, 1985.

The amendment revises the Technical Specifications to 1) reflect shift staffing levels for licensed operators consistent with the provisions of the recently revised 10 CFR 50.54; 2) provide corrections which are typographical or clerical in nature; 3) delete from the Technical Specifications pages referring to out-of-date testing provisions; 4) change an organization chart to reflect a recent organizational change in the offsite engineering support organization; and 5) revise the setting for low condensate storage tank level from "2-inches" to "3%" which is a physically equivalent value. The change is necessitated by the replacement of float type limit switches with analog instruments, with corresponding different units of calibration.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Vernon L. Rooney, Project Manager
Operating Reactors Branch #2
Division of Licensing



Enclosures:
1. Amendment No. 90 to
   License No. DPR-28
2. Safety Evaluation

cc w/enclosure:
See next page

| DISTRIBUTION | | | |
|---|---|---|---|
| Docket File | SNorris | BGrimes | OPA, CMiles |
| NRC PDR | VRooney | TBarnhart (4) | RDiggs |
| Local PDR | OELD | WJones | Gray File |
| ORB#2 Reading | LJHarmon | MVirgilio | Extra - 5 |
| HThompson | ELJordan | ACRS (10) | JPartlow |
| | RHermann | KJohnston | HASchoppman |

| DL:ORB#2 | DL:ORB#2 | DL:ORB#2 | DL:ORB#2 | OELD | DL:ORB#2 | DL:AD-OR |
|---|---|---|---|---|---|---|
| SNorris:ajs | RHermann | VRooney | DVassallo | | KJohnston | GLainas |
| 09/5/85 | 09/10/85 | 09/10/85 | 09/27/85 | 09/30/85 | 09/6/85 | 10/8/85 |

Mr. R. W. Capstick
Vermont Yankee Nuclear Power Corporation
Vermont Yankee Nuclear Power Station

cc:

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 90
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated March 27, 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-28 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 90 , 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

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 9, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 90

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

5
5a
5b
6
6a
19
21a
38
41
47
48
49
51
110f
176
180a
180b
180b1 deleted
180-01
187i
190
191
192
193

## 1.1 SAFETY LIMIT

### 1.1 FUEL CLADDING INTEGRITY

**Applicability**

Applies to the interrelated variable associated with fuel thermal behavior.

**Objective**

To establish limits below which the integrity of the fuel cladding is preserved.

**Specification**

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and core flow is >10% of rated, the existence of a minimum critical power ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

## 2.1 LIMITING SAFETY SYSTEM SETTING

### 2.1 FUEL CLADDING INTEGRITY

**Applicability**

Applies to trip setting of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

**Objective**

To define the level of the process variable at which automatic protective action is initiated.

**Specification**

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

$$S \leq 0.66W + 54\%$$

## 1. SAFETY LIMIT

## 2.1 LIMITING SAFETY SYSTEM SETTING

where:

S = setting in percent of rated thermal power (1593 MWt).

W = percent rated drive flow where 100% rated drive flow is that flow equivalent to $48 \times 10^6$ lbs/hr core flow.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio: $\frac{MFLPD}{FRP}$

where: MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 kW/ft for 8 x 8 fuel.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

1. SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

b. Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, average power range monitor (APRM) scram shall be set down to less than or equal to 15% of rated neutron flux (except as allowed by Note 12 of Table 3.1.1). The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

## 1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure ≤800 psia or Core Flow ≤ of Rated)

When the reactor pressure is ≤800 psia or core flow ≤10% or rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the safety limit established in Specification 1.1A and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The safety limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

## 2.1 LIMITING SAFETY SYSTEM SETTING

B. APRM Rod Block Trip Setting

1. The APRM rod block trip setting shall be as shown in Figure 2.1.1 and shall be:

$$S_{RB} \leq 0.66W + 42\%$$

where:

$S_{RB}$ = rod block setting in percent of rated thermal power (1593 MWt).

W = percent rated drive flow where 100% rated drive flow is that flow equivalent to $48 \times 10^6$ lbs/hr core flow.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio: $\frac{\text{MFLPD}}{\text{FRP}}$

where: MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 kW/ft for 8 x 8 fuel.

FRP = fraction of rated power (1593 MWt).

## 1.1 SAFETY LIMIT

## 2.1 LIMITING SAFETY SYSTEM SETTING

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

## TABLE 3.1.1

## REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

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

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. A turbine stop valve closure and generator load rejection bypass is permitted when the first stage turbine pressure is less than 30% of normal (220 psia).

11. The IRM scram is bypassed when the APRMs are on scale and the mode switch is in the run position.

12. While performing refuel interlock checks which require the mode switch to be in Startup, the reduced APRM high flux scram need not be operable provided:

    a. The following trip functions are operable:

        1. Mode switch in shutdown,
        2. Manual scram,
        3. High flux IRM scram
        4. High flux SRM scram in noncoincidence,
        5. Scram discharge volume high water level, and;

    b. No more than two (2) control rods withdrawn. The two (2) control rods that can be withdrawn cannot be faced adjacent or diagonally adjacent.

TABLE 3.2.1 (Cont)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

High Pressure Coolant Injection System

| Minimum Number of Operable Instrument Channels per Trip System | Trip Function | Trip Level Setting | Required Action When Minimum Conditions for Operation are Not Satisfied |
|---|---|---|---|
| 2 (Note 3) | Low-Low Reactor Vessel Water Level | Same as LPCI | Note 5 |
| 2 (Note 4) | Low Condensate Storage Tank Water Level | ≥ 3% | Note 5 |
| 2 (Note 3) | High Drywell Pressure | Same as LPCI | Note 5 |
| 1 (Note 3) | Bus Power Monitor | -- | Note 5 |
| 1 (Note 4) | Trip System Logic | -- | Note 5 |
| 2 (Note 7) | High Reactor Vessel Water Level | ≤177 inches above top of enriched fuel | Note 5 |



Amendment No. ~~68~~, ~~85~~ 90

TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

| Minimum Number of Operable Instrument Channels per Trip System | Trip Function | Trip Setting | Required Action When Minimum Conditions for Operations are Not Satisfied (Note 2) |
|---|---|---|---|
| 2 | Low-Low Reactor Vessel Water Level | $\geq$82.5" above the top of enriched fuel | A |
| 2 of 4 in each of 2 channels | High Main Steam Line Area Temperature | $\leq$212°F | B |
| 2/steam line | High Main Steam Line Flow | $\leq$140% of rated flow | B |
| 2/(Note 1) | Low Main Steam Line Pressure | $\geq$800 psig | B |
| 2/(Note 6) | High Main Steam Line Flow | $\leq$40% of rated flow | B |
| 2 | Low Reactor Vessel Water Level | Same as Reactor Protection System | A |
| 2 | High Main Steam Line Radiation (7) (8) | $\leq$3 X background at rated power (9) | B |
| 2 | High Drywell Pressure | Same as Reactor Protection System | A |
| 2/(Note 10) | Condenser Low Vacuum | $\leq$12" Hg absolute | A |
| 1 | Trip System Logic | -- | A |

Amendment No. ~~68~~, ~~84~~, ~~86~~ 90

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

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

TABLE 3.2.5 NOTES

1. There shall be two operable or tripped trip systems for each function in the required operating mode. If the minimum number of operable instruments are not available for one of the two trip systems, this condition may exist for up to seven days provided that during the time the operable system is functionally tested immediately and daily thereafter; if the condition lasts longer than seven days, the system shall be tripped. If the minimum number of instrument channels are not available for both trip systems, the systems shall be tripped.

2. One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

3. This function may be bypassed when count rate is $\geq$100 cps or when all IRM range switches are above Position 2.

4. IRM downscale may be bypassed when it is on its lowest scale.

5. "W" is percent rated drive flow where 100% rated drive flow is that flow equivalent to 48 x $10^6$ lbs/hr core flow. Refer to L.C.O. 3.11.C for acceptable values for N.

6. The minimum number of operable instrument channels may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30-day period.

7. The trip may be bypassed when the reactor power is $\leq$30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

8. With the number of operable channels less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition without one hour.

9. With one RBM channel inoperable:

   a. Verify that the reactor is not operating on a limiting control rod pattern, and

   b. Restore the inoperable RBM channel to operable status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

TABLE 3.2.6

POST-ACCIDENT INSTRUMENTATION

| Minimum Number of Operable Instrument Channels | Parameter | Type of Indication | Instrument Range |
|---|---|---|---|
| 2 | Drywell Atmospheric Temperature (Note 1) | Recorder #16-19-45<br>Recorder #TR-1-149 | 0-300$^{o}$F<br>0-300$^{o}$F |
| 2 | Drywell Pressure (Note 1)<br>Torus Pressure (Note 1) | Recorder #16-19-44 | 0-80 psia<br>0-80 psia |
| 2 | Torus Water Level (Note 3) | Meter #16-19-46A<br>Meter #16-19-46B | 0-3 ft.<br>0-3 ft. |
| 2 | Torus Water Temperature (Note 1) | Meter #16-19-48 | 60-180$^{o}$F |
| 2 | Reactor Pressure (Note 1) | Recorder #6-97<br>Meter #6-90A<br>Meter #6-90B | 0-1200 psig<br>0-1200 psig<br>0-1200 psig |
| 2 | Reactor Vessel Water Level (Note 1) | Meter #2-3-91A<br>Meter #2-3-91B | (-150)-0-(+150)"$H_2O$<br>(-150)-0-(+150)"$H_2O$ |
| 1 | Control Rod Position (Note 1,2) | Meter | 0-48" RPIS |
| 1 | Neutron Monitor (Note 1,2) | Meter | 0-125% Rated flux |
| 1 | Torus Air Temperature (Note 1) | Recorder #TR-16-19-45 | 0-300$^{o}$F |
| 2/valve | Safety/Relief Valve Position from pressure switches (Note 4) | Lights (SRV 2-71-A through D) | Closed - Open |
| 1/valve | Safety Valve Position from Acoustic Monitor (Note 5) | Meter Z1-2-1A/B | Closed - Open |

Note 1 - From and after the date that one of these parameters is not indicated in the Control Room, continued reactor operation is permissible during the next seven days. If reduced to one indication of a parameter operation is permissible for 30 days.

Note 2 - Control rod position and neutron monitor instruments are considered to be redundant to each other.

TABLE 4.2.1 (Cont)

| Low Pressure Coolant Injection System | | | |
|---|---|---|---|
| **Trip Function** | **Function Test (8)** | **Calibration (8)** | **Instrument Check** |
| Low Reactor Pressure #1 | (Note 1) | Once/Operating Cycle | --- |
| High Drywell Pressure #1 | (Note 1) | Once/Operating Cycle | Once Each Day |
| Low-Low Reactor Vessel Water Level | (Note 1) | Once/Operating Cycle | Once Each Day |
| Reactor Vessel Shroud Level | (Note 1) | Every 3 Months | --- |
| Low Reactor Pressure #2 | (Note 1) | Every 3 Months | --- |
| RHR Pump Discharge Pressure | (Note 1) | Every 3 Months | --- |
| High Drywell Pressure #2 | (Note 1) | Every 3 Months | --- |
| Low Reactor Pressure #3 | (Note 1) | Once/Operating Cycle | --- |
| Auxiliary Power Monitor | (Note 1) | Every Refueling Outage | Once Each Day |
| Pump Bus Power Monitor | (Note 1) | None | Once Each Day |
| LPCI Crosstie Monitor | None | None | Once Each Day |
| Trip System Logic | Every 6 Months (Note 2) | Every 6 Months (Note 3) | --- |

## 3.6 LIMITING CONDITIONS FOR OPERATION

### J. Thermal Hydraulic Stability

When the reactor mode switch is in RUN, the reactor shall not intentionally be operated in a natural circulation mode nor shall an idle recirculation pump be started with the reactor in a natural circulation mode.

## 4.6 SURVEILLANCE REQUIREMENTS

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

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

### J. Thermal Hydraulic Stability

## 3.10 LIMITING CONDITIONS FOR OPERATION

### B. Operation with Inoperable Components

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.10.A shall be met except:

1. **Diesel Generators**

   From and after the date that one of the diesel generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel generator is operable, the requirements of Specification 3.5.H.1 shall be satisfied.

2. **Batteries**

   a. From and after the date that ventilation is lost in the battery room portable ventilation equipment shall be provided.

   b. From and after the date that one of the two 125 volt station battery systems is made or found to be inoperable for any reasons, continued reactor operation is permissible only during the succeeding three days provided Specification 3.5.H is met unless such battery system is sooner made operable.

   c. From and after the date that one of the two 24 volt ECCS Instrumentation battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding three days unless such battery system is sooner made operable.

## 4.10 SURVEILLANCE REQUIREMENTS

### B. Operation with Inoperable Components

1. **Diesel Generators**

   When it is determined that one of the diesel generators is inoperable the requirements of Specification 4.5.H.1 shall be satisfied.

2. **Batteries**

   Samples of the Battery Room atmosphere shall be taken daily for hydrogen concentration determination.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting values shown in Tables 3.11-1A through G. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at >25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

prescribed limits within two (2) hours, the reactor shall be brought to the shutdown conditions within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

B. <u>Linear Heat Generation Rate (LHGR)</u>

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR of 13.4 kW/ft for 8x8, 8x8R, and P8x8R.

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits with two (2) hours, the reactor shall be brought to shutdown condition within 36 hours, Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

B. <u>Linear Heat Generation Rate (LHGR)</u>

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

C. <u>Minimum Critical Power Ratio</u>

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.6.

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VYNPS
Table 3.11-2
MCPR Operating Limits

| Value of "N" in RBM Equation (1) | Average Control Rod Scram Time | Cycle Exposure Range | MCPR Operating Limit for Fuel Type (2) | | |
|---|---|---|---|---|---|
| | | | 8x8 | 8x8R | P8x8R |
| 42% | Equal or better than L.C.O. 3.3 C.1.1 | BOC to EOC-2 GWD/T | 1.29 | 1.29 | 1.29 |
| | | EOC-2 GWD/T to EOC-1 GWD/T | 1.29 | 1.29 | 1.29 |
| | | EOC-1 GWD/T to EOC | 1.30 | 1.30 | 1.30 |
| | Equal or better than L.C.O. 3.3 C.1.2 | BOC to EOC-2 GWD/T | 1.29 | 1.29 | 1.29 |
| | | EOC-2 GWD/T to EOC-1 GWD/T | 1.33 | 1.31 | 1.31 |
| | | EOC-1 GWD/T to EOC | 1.36 | 1.35 | 1.35 |
| 41% | Equal or better than L.C.O. 3.3 C.1.1 | BOC to EOC-2 GWD/T | 1.25 | 1.25 | 1.25 |
| | | EOC-2 GWD/T to EOC-1 GWD/T | 1.26 | 1.25 | 1.25 |
| | | EOC-1 GWD/T to EOC | 1.30 | 1.30 | 1.30 |
| | Equal or better than L.C.O. 3.3 C.1.2 | BOC to EOC-2 GWD/T | 1.25 | 1.25 | 1.25 |
| | | EOC-2 GWD/T to EOC-1 GWD/T | 1.33 | 1.31 | 1.31 |
| | | EOC-1 GWD/T to EOC | 1.36 | 1.35 | 1.35 |
| ≤ 40% | Equal or better than L.C.O. 3.3 C.1.1 | BOC to EOC-2 GWD/T | 1.25 | 1.25 | 1.25 |
| | | EOC-2 GWD/T to EOC-1 GWD/T | 1.26 | 1.25 | 1.25 |
| | | EOC-1 GWD/T to EOC | 1.30 | 1.30 | 1.30 |
| | Equal or better than L.C.O. 3.3 C.1.2 | BOC to EOC-2 GWD/T | 1.25 | 1.25 | 1.25 |
| | | EOC-2 GWD/T to EOC-1 GWD/T | 1.33 | 1.31 | 1.31 |
| | | EOC-1 GWD/T to EOC | 1.36 | 1.35 | 1.35 |

(1) The Rod Block Monitor (RBM) trip setpoints are determined by the equation shown in Table 3.2.5 of the Technical Specifications.

(2) The current analyses for MCPR Operating Limits do not include 7x7 fuel. On this basis, further evaluation of MCPR Operating Limits is required before 7x7 fuel can be used in reactor power operation.

3.13 LIMITING CONDITIONS FOR OPERATION

3. Except as specified in Specification 3.13.G.4 below, the Turbine Building Foam System shall be operable with its foam concentrate tank full (150 gallons).

4. From and after the date that the Turbine Building Foam System is inoperable a portable foam nozzle shall be brought to the Turbine Building Foam System location. A 150 gallon foam concentrate supply shall be available on-site.

4.13 SURVEILLANCE REQUIREMENTS

1. Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

2. A visual inspection of the foam system and equipment to verify integrity, and

3. A visual inspection of the Recirculation M.G. Set Foam System foam nozzle area to verify that the spray pattern is not obstructed.

4. Foam concentrate samples shall be taken and analyzed for acceptability.

d. At least once per 3 years by performing an air flow test through the Recirculation M.G. Set foam header and verifying each foam nozzle is unobstructed.

## 6.0 ADMINISTRATIVE CONTROLS

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

## 6.1 ORGANIZATION

A. The Plant Manager has on-site responsibility for the safety and efficient operation of the facility. Succession to this responsibility during his absence shall be delegated in writing.

B. The portion of the corporate management which relates to the operation of this plant is shown in Figure 6.1.1.

C. In all matters relating to the operation of the plant and to those Technical Specifications, the Plant Manager shall report to and be directly responsible to the Manager of Operations.

D. Conduct of operations of the plant is shown in Figure 6.1.2 and will be in accordance with the following minimum conditions.

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

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

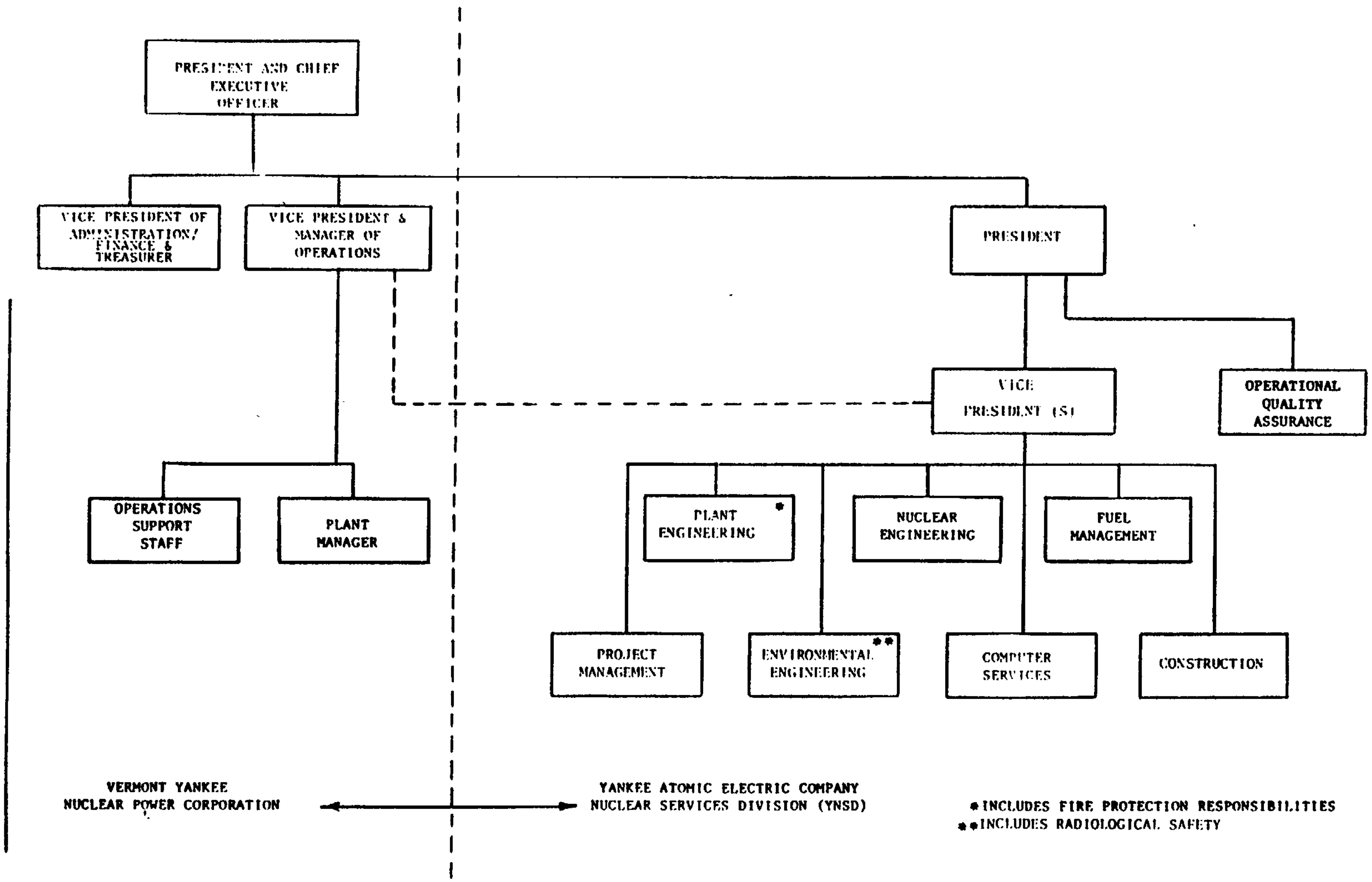
3. A dedicated, licensed Senior Operator shall be in charge of any reactor core alteration.

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

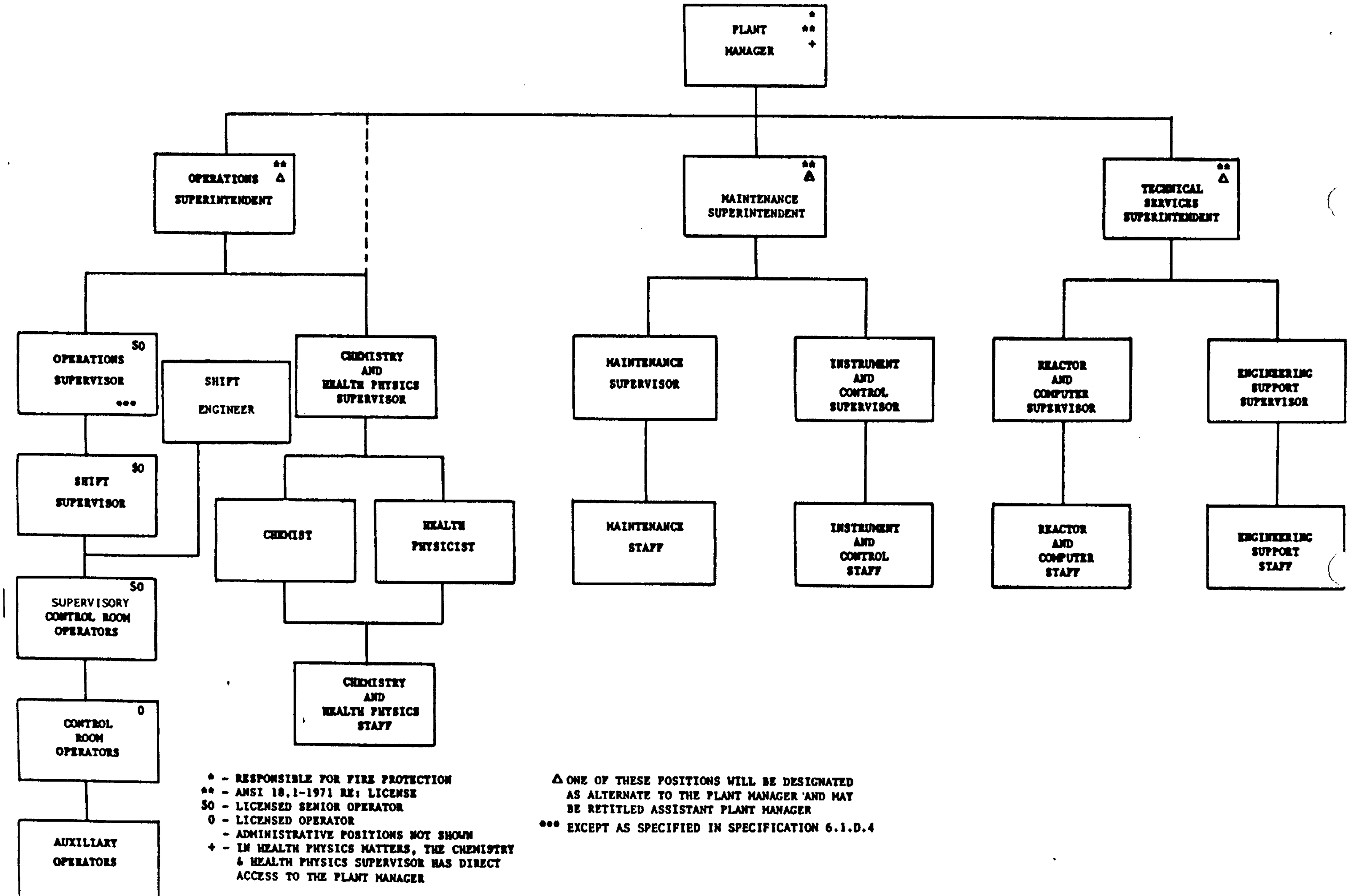
   a. Plant Manager
   b. Operations Superintendent
   c. Technical Services Superintendent
   d. Maintenance Superintendent
   e. Chemistry and Health Physics Supervisor
   f. Operations Supervisor (See Item D.7, Page 190a)
   g. Reactor and Computer Supervisor
   h. Maintenance Supervisor
   i. Instrument and Control Supervisor
   j. Shift Supervisors

FIGURE 6.1.1

OFFSITE SUPPORT ORGANIZATION



VERMONT YANKEE NUCLEAR POWER STATION




* - RESPONSIBLE FOR FIRE PROTECTION
** - ANSI 18.1-1971 RE: LICENSE
SO - LICENSED SENIOR OPERATOR
O - LICENSED OPERATOR
- ADMINISTRATIVE POSITIONS NOT SHOWN
\+ - IN HEALTH PHYSICS MATTERS, THE CHEMISTRY & HEALTH PHYSICS SUPERVISOR HAS DIRECT ACCESS TO THE PLANT MANAGER

Δ ONE OF THESE POSITIONS WILL BE DESIGNATED AS ALTERNATE TO THE PLANT MANAGER AND MAY BE RETITLED ASSISTANT PLANT MANAGER
*** EXCEPT AS SPECIFIED IN SPECIFICATION 6.1.D.4

Amendment No. ~~66~~, ~~75~~, ~~79~~, ~~80~~, ~~87~~ 90

TABLE 6.1.1

Vermont Yankee staff positions that shall be filled by personnel holding Senior Operator and Operator licenses are indicated in the following table:

| Title | License Requirements |
|---|---|
| Operations Supervisor | Licensed Senior Operator (Except as specified in Specification 6.1.D.4) |
| Shift Supervisor | Licensed Senior Operator |
| Supervisory Control Room Operator | Licensed Senior Operator |
| Control Room Operator | Licensed Operator |

| MINIMUM SHIFT STAFFING ON-SITE | Conditions | |
|---|---|---|
| | Plant Startup and Normal Operation (Note 1) | Cold Shutdown or Refueling with Fuel in the Reactor (Note 2) |
| Shift Supervisor | 1 | 1 |
| Supervisory Control Room Operator | 1 | - |
| Control Room Operator | 2 | 1 |
| Auxiliary Operator | 2 | 1 |
| Shift Engineer | 1 | - |

**Notes**

(1) At least one Senior Licensed Operator and one Licensed Operator, <u>or</u> two Senior Licensed Operators, shall be in the Control Room.

(2) At least one Licensed Operator, <u>or</u> one Senior Licensed Operator, shall be in the Control Room.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 90 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

## 1.0 INTRODUCTION

By letter dated March 27, 1985, Vermont Yankee Nuclear Power Corporation (the licensee) requested an amendment to the Technical Specifications, Appendix A, Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment requested changes to the Technical Specifications to 1) reflect shift staffing levels for licensed operators consistent with the provisions of the recently revised 10 CFR 50.54; 2) provide corrections which are typographical or clerical in nature; 3) delete from the Technical Specifications pages referring to out-of-date testing provisions; 4) change an organization chart to reflect a recent organizational change in the offsite engineering support organization; and 5) revise the setting for low condensate storage tank level from "2-inches" to "3%" which is a physically equivalent value. The change is necessitated by the replacement of float type limit switches with analog instruments, with corresponding different units of calibration.

## 2.0 EVALUATION

### 2.1 Shift Staffing Levels

The existing Specification 6.1.D contains a parenthetical reference to Table 6.1.1. The revised Specification 6.1.D deletes this reference. This is considered an editorial change and is acceptable. Table 6.1.1 is adequately referenced in revised Specification 6.1.D.2.

The existing Specification 6.1.D.1 requires a licensed senior operator and an individual qualified in radiation protection procedures to be on site whenever there is fuel in the reactor. The revised Specification 6.1.D.1 deletes the words associated with the licensed senior operator. This is acceptable because the requirement for having a senior operator on site whenever there is fuel in the reactor is covered by revised Table 6.1.1.

The existing Specification 6.1.D.2 requires that "Licensed Operators on site shall be in accordance with Table 6.1.1," and it requires at least one licensed operator to be in the control room whenever there is fuel in the reactor. The revised Specification 6.1.D.2 changes the wording somewhat by requiring the "minimum shift staffing on site shall be in accordance with Table 6.1.1," and it deletes the words associated with a licensed operator

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having to be in the control room. The first change is considered to be an editorial change and is acceptable. The second change is also acceptable because the requirement for having a licensed operator in the control room whenever there is fuel in the reactor is covered by revised Table 6.1.1.

The existing Specification 6.1.D.3 requires that "a Licensed Senior Operator shall be in charge of any refueling operation." The revised Specification 6.1.D.3 requires that "a dedicated, Licensed Senior Operator shall be in charge of any reactor core alteration." This change brings the wording of Specification 6.1.D.3 into closer agreement with 10 CFR 50.54(m)(2)(iv) and is acceptable.

The existing Figure 6.1.2 (on-site organization chart) has a box in the operations chain of command containing the title "Senior Control Room Operators." The revised Figure 6.1.2 uses the title "Supervisory Control Room Operators," which brings the title used in the figure into agreement with the title used in Table 6.1.1. This is an editorial change and is acceptable.

The existing Table 6.1.1 establishes operator license and shift staffing requirements for the Vermont Yankee plant. The table has been revised to conform with 10 CFR 50.54(m)(2); several editorial changes have also been made. The proposed changes, listed below, are acceptable:

- The column entitled "License" has been changed to "License Requirements," and the column entries entitled "Senior Operator" and "Operator" have been changed to "Licensed Senior Operator" and "Licensed Operator," respectively. These are editorial changes.

- The license requirement for the Supervisory Control Room Operator has been upgraded from "Operator" to "Licensed Senior Operator." This brings the table into conformance with 10 CFR 50.54(m)(2)(i) by requiring two senior operators on each shift during startup and normal operation (the Shift Supervisor and the Supervisory Control Room Operator).

- The column entitled "Minimum Shift Crew Personnel & License Requirements" has been changed to "Minimum Shift Staffing On-Site." This is considered an editorial change, since license requirements are specified elsewhere on the table.

- The existing columns entitled "Normal Operation" and "Plant Startup" are identical, therefore, they have been combined into a single column entitled "Plant Startup and Normal Operation." This is considered an editorial change.

- The column entitled "Cold Shutdown" has been changed to "Cold Shutdown or Refueling with Fuel in the Reactor." By making this column apply in refueling situations with fuel in the reactor, the licensee can accommodate the proposed deletion from Specification 6.1.D.1 described above.

- o In the revised column entitled "Cold Shutdown or Refueling with Fuel in the Reactor," the entry specifying the minimum number of Shift Supervisors has been changed from 0 to 1, and the entry specifying the minimum number of Supervisory Control Room Operators has been changed from 1 to 0. This change upgrades the table by requiring the senior member of the shift crew to be on site when the plant is in cold shutdown or refueling (with fuel in the reactor). The proposed change conforms with 10 CFR 50.54(m)(2)(ii).

- o The existing table contains summary information on the total number of operator licenses and senior operator licenses required on shift under various operating conditions. This information is redundant to other information in the table and has been deleted. This is considered an editorial change.

- o The revised table contains two new footnotes that specify the number of licensed operators and/or senior licensed operators required to be in the control room under various operating conditions. By making this change, the licensee can accommodate the proposed deletion from Specification 6.1.D.2 described above. The proposed change conforms with 10 CFR 50.54(m)(2)(iii).

2.2 Editorial Changes

The licensee has requested the following revisions to pages 5b, 41, 49, 51, 176 and 187i of the Technical Specification to correct typographical errors:

The change to page 5b corrects the reference to Note number for Table 3.1.1 from 13 to 12.

The change to page 41 (Table 3.2.2) corrects the acceptable low vacuum setpoint from "greater than or equal to 12" Hg absolute to "less than or equal to 12" Hg absolute. This error was identified in I&E Inspection Report No. 50-271/83-17, dated August 3, 1983.

The change to page 49 adds a reference to Note 1 for torus water temperature which was inadvertently deleted by the issuance of Amendment No. 63 to the Facility Operating License.

The change to page 51 corrects the calibration requirements for the Low Reactor Pressure Trip Functions #2 and #3.

The change to page 176 corrects Specification 3.10.B to refer to Specification 3.10.A rather than 3.9.A.

The change to page 187i corrects Specification 3.13.G.4 to refer to foam system being "inoperable" rather than "operable."

We have reviewed the changes, find they are editorial in nature and, therefore, are acceptable.

2.3 Deletion of Out-of-date Testing Provisions

By letter dated February 21, 1981 Vermont Yankee requested a change to the Technical Specifications Table 3.2.1 to permit stability and recirculation pump trip tests to be performed at the facility during Cycle 8. The staff issued Amendment No. 64 on March 11, 1981 permitting the requested tests. The actual time stated to perform the tests was 48 hours over a period of 7 days.

The licensee, as a part of its March 27, 1985 request, asked that the Technical Specifications be changed to delete the test provisions since the plant has completed Cycle 8 and these provisions are no longer needed. We find the change acceptable because it is basically editorial in nature.

2.4 Offsite Engineering Support Organization

Figure 6.1.1 is revised to update the organization of the Nuclear Services Division of the Yankee Atomic Electric Company, which provides offsite engineering support to the Vermont Yankee plant. The "Vice President--Operations" position is deleted, and the Plant Engineering function that formerly reported to that position is moved under the remaining "Vice Président(s)" positions.

In addition, the "Radiological Safety" box on existing Figure 6.1.1 is deleted and a footnote is added to show that the "Environmental Engineering" function includes radiological safety.

Finally, a new "Project Management" box is added to Figure 6.1.1.

The proposed changes described above do not affect the scope of offsite engineering support provided by the Yankee Atomic Electric Company. The changes more accurately represent the offsite engineering support organization and are acceptable.

2.5 Condensate Storage Tank Low Water Level Trip

Table 3.2.1, "Emergency Core Cooling System Actuation Instrumentation," is revised to reflect a new trip level setting for the Condensate Storage Tank (CST) water level. This change is necessitated by the replacement of the original float level switches with analog instruments which have different units of calibration. The setting for low CST level is revised from "2-inches" to "3%." This is effectively the same trip level setting presently required by the Technical Specifications, and is, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

## 4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: October 9, 1985