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Docket No. 50-271

MAY 21 1975

Yankee Atomic Electric Company
ATTN: Mr. G. Carl Andognini
Assistant to the Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 14 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment includes Change No. 25 to the Technical Specifications and is in response to Vermont Yankee's request dated March 20, 1975.

This amendment incorporates corrections necessitated by previous license amendments and organizational changes which have occurred at the corporate and operational levels. During our review, we discussed with your staff certain modifications to the proposed changes which they agreed were necessary for clarification. These modifications have been made.

Copies of our Safety Evaluation and the Federal Register Notice relating to this action are also enclosed.

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 14
   w/Change No. 25
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

bcc: J. R. Buchanan, ORNL
T. B. Abernathy, DTIE

| | | | | | | |
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| OFFICE ▶ | RL:ORB-2 x7403 | RL:ORB-2 | OELD | RL:ORB-2 | | |
| SURNAME ▶ | RMDiggs | FDAnderson | DEKartalia | DLZiemann | | |
| DATE ▶ | 4/3/75 | 4/4/75 | 4/21/75 | 5/21/75 | | |

Form AEC-318 (Rev. 9-53) AECM 0240 GPO 643-16-81465-1 445-678

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VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
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 20, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-28 is hereby amended to read as follows:

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"B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 25".

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Attachment:
Change No. 25 to the
Technical Specifications

Date of Issuance: MAY 21 1975

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| OFFICE ▶ | | | | | | |
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ATTACHMENT TO LICENSE AMENDMENT NO. 14

CHANGE NO. 25 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Delete pages 21, 23, 25, 46, 51, 72, 110, 118R, 123, 124, 125, 142, 191, 192 and 194 from the Appendix A Technical Specifications and insert the same numbered attached replacement pages. The changed areas on the revised pages are shown by marginal lines.

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TABLE 3.1.1 NOTES

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

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

2. There shall be two operable or tripped trip systems for each function.

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

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   B. Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   C. Reduce turbine load and close main steamline isolation valves within eight hours.
   D. Reduce reactor power to less than 30% of rated within eight hours.

4. "W" is the reactor driving loop flow in percent of rated.

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

6. 1 inch on the water level instrumentation is 127 above the top of the active fuel.

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

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

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. Fast closure system equipment includes acceleration-relay pressure switches, APRM setdown timers, generator load rejection timers, and associated relays. 25

10. A turbine stop valve closure and generator load rejection bypass is permitted when the first stage turbine pressure is less than 30 percent 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.

TABLE 4.1.1 (CONT'D)

| Instrument Channel | Group (3) | Functional Test (7) | Minimum Frequency (4) |
|---|---|---|---|
| Scram Test Switch | A | Trip Channel and Alarm | Each Refueling Outage |
| First Stage Turbine Pressure - Permissive | A | Trip Channel and Alarm | Every 6 Months |

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TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

| Instrument Channel | Group (1) | Calibration Standard (5) | Minimum Frequency (2) |
|---|---|---|---|
| High Flux IRM | C | Comparison to APRM after Heat Balance | Every controlled shutdown (4) |
| High Flux APRM | | | |
| Output Signal | B | Heat Balance | Once Every 7 Days |
| Flow Bias | B | Standard Pressure and Voltage Source | Refueling Outage |
| LPRM | B (6) | Using TIP System | Every 1000 equiv. full pwr. hr. |
| High Reactor Pressure | A | Standard Pressure Source | Every 3 months |
| Turbine Control Valve Fast Closure | A | Standard Pressure Source | Every 3 months |
| High Drywell Pressure | A | Standard Pressure Source | Every 3 months |
| High Water Level in Scram Discharge Volume | A | Water Level | Refueling Outage |
| Low Reactor Water Level | A | Water Level | Every 3 months |
| Turbine Stop Valve Closure | A | (7) | Refueling Outage |
| High Main Steamline Radiation | B | Appropriate Radiation Source (3) | Refueling Outage |
| First Stage Turbine Pressure - Permissive | A | Pressure Source | Every 6 months and after refueling |
| Main Steamline Isolation Valve Closure | A | (7) | Refueling Outage |

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TABLE 3.2.4

OFF GAS SYSTEM ISOLATION INSTRUMENTATION

| Minimum Number of Operable Instrument Channels per Trip System | Trip Function | Trip Setting | Required Action When Minimum Condition for Operation are not Met |
|---|---|---|---|
| 2 | Air Ejector Off-Gas Radiation | $\leq$ 1.5 Ci/sec (3)<br>$\leq$ 0.3 Ci/sec (4) | Notes 1 & 2 |
| 1 | Time Delay (Air Ejector Suction Valve Isolation)(17-157&17-157A) | $\leq$ 15 minutes<br>$\leq$ 1 minute | Notes 1 & 2 |
| 1 | Logic Bus Power Monitor | --- | Notes 1 & 2 |
| 1 | Trip System Logic | --- | Notes 1 & 2 |
| 2 | Augmented Off-Gas Radiation | $\leq$ 0.07 Ci/sec (6)(7) | Note 5 |
| 1 | Time Delay (Stack Off-Gas Valve Isolation) (15TD & 16TD) | $\leq$ 2 minutes<br>$\leq$ 30 minutes | Note 5 |
| 1 | Trip System Logic | --- | Note 5 |

Note 1 - If the minimum number of operable instrument channels are not available, reactor power operation is permissible for only seven successive days unless the system is sooner made operable. 25

Note 2 - If the trip system is inoperable or if both instrument channels are unavailable, initiate an orderly shutdown and have the reactor in the cold shutdown condition within 24 hours.

Note 3 - If the radiation level exceeds 1.5 Ci/sec (30 minute decay level), the air ejector suction valves shall close within one minute.

Note 4 - If the radiation level exceeds 0.3 Ci/sec (30 minute decay level), the air ejector suction valves shall close unless the radiation level decreases to less than 0.3 Ci/sec within 15 minutes.

Note 5 - At least one of the radiation monitors between the charcoal bed system and the plant stack shall be operable during operation of the augmented off-gas system. If this condition cannot be met, continued operation of the augmented off-gas system is permissible for a period of up to 7 days provided that at least one of the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

Note 6 - If the radiation level exceeds 0.07 Ci/sec, the stack isolation valve shall close within 2 minutes if the dryer-absorber system bypass valves (OG-145 & -146) are open and within 30 minutes if these valves are closed unless the radiation level has decreased below the trip point sooner.

Note 7 - During plant startup or shutdown conditions (changing disintegration energy; $\overline{E}\gamma$) the trip point may be adjusted upward but must always remain below a rate corresponding to $0.08/\overline{E}\gamma$.

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) | every 3 months | ---- | |
| High Drywell Pressure #1 | (Note 1) | every 3 months | ---- | |
| Low-Low Reactor Vessel Water Level | (Note 1) | every 3 months | 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) | every 3 months | ---- | |
| Auxiliary Power Monitor | (Note 1) | every refueling outage | once each day | 25 |
| Pump Bus Power Monitor | (Note 1) | None | once each day | |
| LPCI Crosstie Monitor | None | None | once each day | 25 |
| Trip System Logic | Every 6 Months (Note 2) | every 6 months (Note 3) | ---- | |

3 LIMITING CONDITIONS FOR OPERATION

Scram Insertion Times

1. The average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

| % Inserted From Fully Withdrawn | Avg. Scram Insertion Times (sec) |
|---|---|
| 5 | 0.375 |
| 20 | 0.90 |
| 50 | 2.00 |
| 90 | 3.50 |

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

| %Inserted From Fully Withdrawn | Avg. Scram Insertion Times (sec) |
|---|---|
| 5 | 0.398 |
| 20 | 0.954 |
| 50 | 2.120 |
| 90 | 3.80 |

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

4.3 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power, with reactor pressure above 800 psig, all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. During or following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the semiannual operating report to the Commission. 25

| 3.6 LIMITING CONDITION FOR OPERATION | 4.6 SURVEILLANCE REQUIREMENT |
| --- | --- |
| | 3. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle. |

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The "worst case" curve relating change in transition temperature to neutron fluence as presented in the FSAR was used to construct the "Minimum Reactor Pressurization Temperature" curve of Figure 3.6.1. This curve is based on an intitial NDT of the vessel shell adjacent to the core. A 60°F margin based on the requirements of Section III of the ASME Boiler and Pressure Vessel Code, and a 60°F margin to account for the thickness effect of heavy section steel were added to 40°F to give the 160°F minimum temperature from initial operation to the time when the neutron fluence exceeds $5 \times 10^{16}$ nvt. At that time the minimum temperature will increase steadily as the neutron fluence increases based on the "worst case" curve. After 40 years of operation the minimum operating temperature will be about 180°F.

The reactor vessel head flange and the vessel flange in combination with the double "O" ring type seal are designed to provide a leak tight seal when bolted together. When the vessel head is placed on the reactor vessel, only that portion of the head flange near the inside of the vessel rests on the vessel flange. As the head bolts are replaced and tensioned, the vessel head is flexed slightly to bring together the entire contact surfaces adjacent to the "O" rings of the head and vessel flange. The head flange and adjacent plate have an NDT of 10°F and are not subjected to any appreciable neutron fluence; therefore, the minimum temperature for bolting the vessel flange is 10°F + 60°F = 70°F.

Numerous data are available relating integrated flux and the change in nil-ductility transition temperature (NDTT) in various steels. The most conservative data has been used in Specification 3.6. The integrated flux at the vessel wall is calculated from core physics data and will be measured using flux monitors installed inside the vessel. The measurements of the neutron flux at the vessel wall will be used to check and, if necessary, correct the calculated data to determine an accurate NDTT.

In addition, vessel material samples will be located within the vessel to monitor the effect of neutron exposure on these materials. The samples include specimens of base metal, weld zone metal, heat affected zone metal, and standard specimens. These samples will receive neutron exposure more rapidly than the vessel wall material and, therefore, will lead the vessel in integrated neutron flux exposure. These samples will provide further assurance that the shift in NDTT used in the specification is conservative.

B. Coolant Chemistry

A steady state radioiodine concentration limit of 1.1 μCi of I-131 dose equivalent per gram of water in the reactor coolant system can be reached if the gross radioactivity in the gaseous effluents are near the limit as set forth in Specification 3.8.C.1.a or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was 25

3.6 & 4.6 (CONT'D)

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

The in-service inspection program presented at this time is based on a thorough evaluation of present technology and state-of-the-art inspection techniques. The program will be continually re-evaluated as technology in the field of non-destructive inspection and equipment development. After five years, a new program will be presented to the NRC.

The interest of Vermont Yankee Nuclear Power Corporation in the development of new techniques for non-destructive testing of nuclear pressure boundaries is indicated by their participation in an Edison Electric Institute sponsored project. This project is primarily aimed at developing new techniques for continuous in-service inspection monitoring; namely, the accoustic emission and accoustic spectrometer techniques. The EEI program is also devoting some funding to the improved conventional ultrasonic inspection techniques.

F. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ±5%, the flow rates in both recirculation loops will be verified by main Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and
25 derived core flow rate is 10% or more (with the measured value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measure. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate.

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3.6 & 4.6 (cont'd)

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Specifications 4.6.F.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus, the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

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4.7.A (CONT'D)

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (La) to a value of 0.80%/day.

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

To allow a margin for possible leakage deterioration between test intervals, the maximum allowable operational leak rate (Ltm), which will be met to remain on the normal test schedule, is 0.75 Lt.
In addition, it is our intent to operate the primary containment structure at a slight positive pressure and to continuously monitor primary containment leakage. During normal plant operation only infrequent gas additions will be made to the primary containment and these will be made manually through a calibrated gas meter. Continuous primary containment temperature, pressure, and relative humidity data will be fed to the computer, which in turn will automatically calculate, by the absolute method, the actual weight of gas within the primary containment. This variation with time is the leak-
age rate and any change in this value is easily seen and permits corrective action to be taken to in- 25
sure that the primary containment integrity is maintained.

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

# WESTBORO OFFICE OF THE YANKEE ATOMIC ELECTRIC COMPANY (NSD)

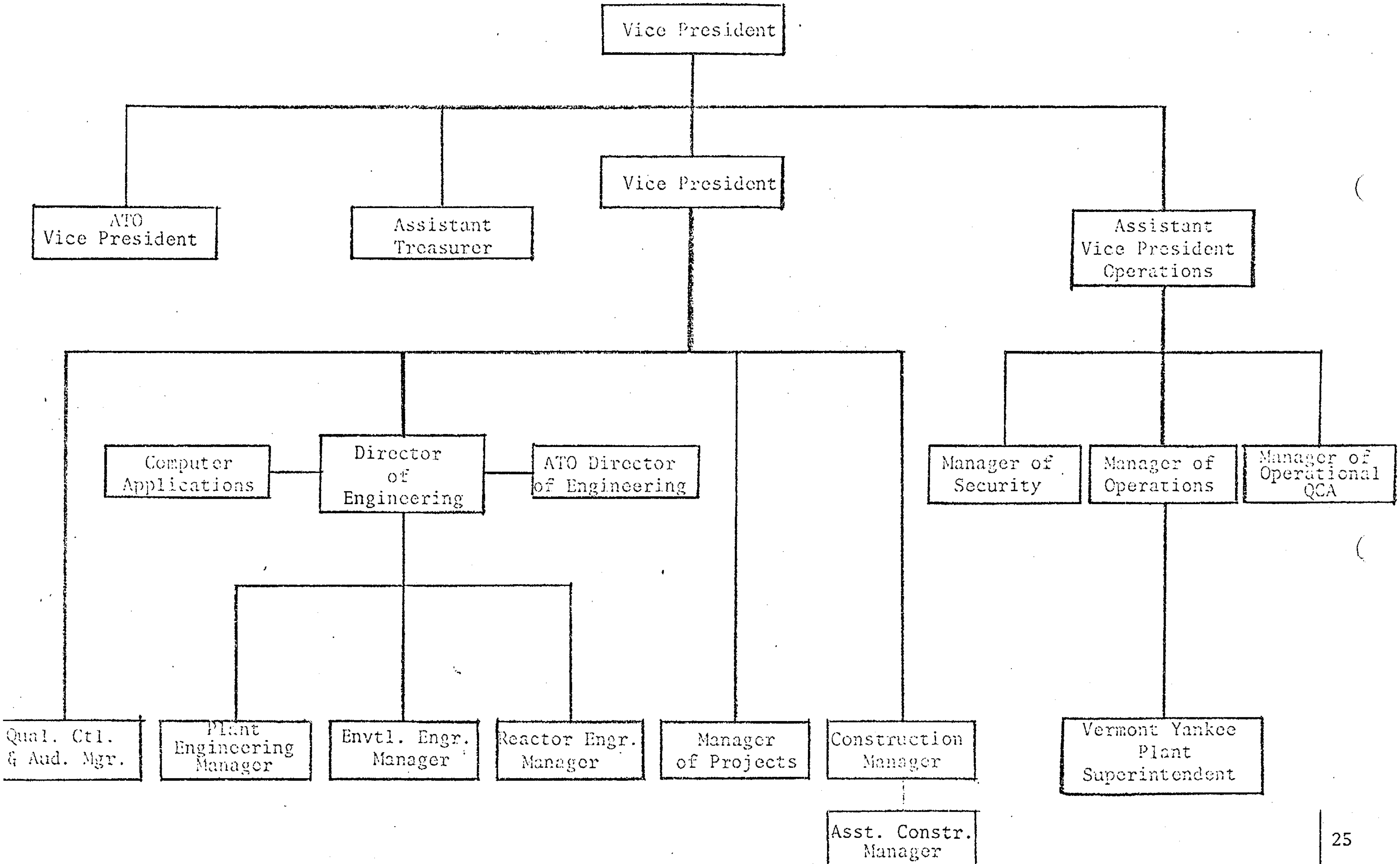


Figure 6.1.1 CORPORATE ORGANIZATION

VERMONT YANKEE NUCLEAR POWER CORPORATION

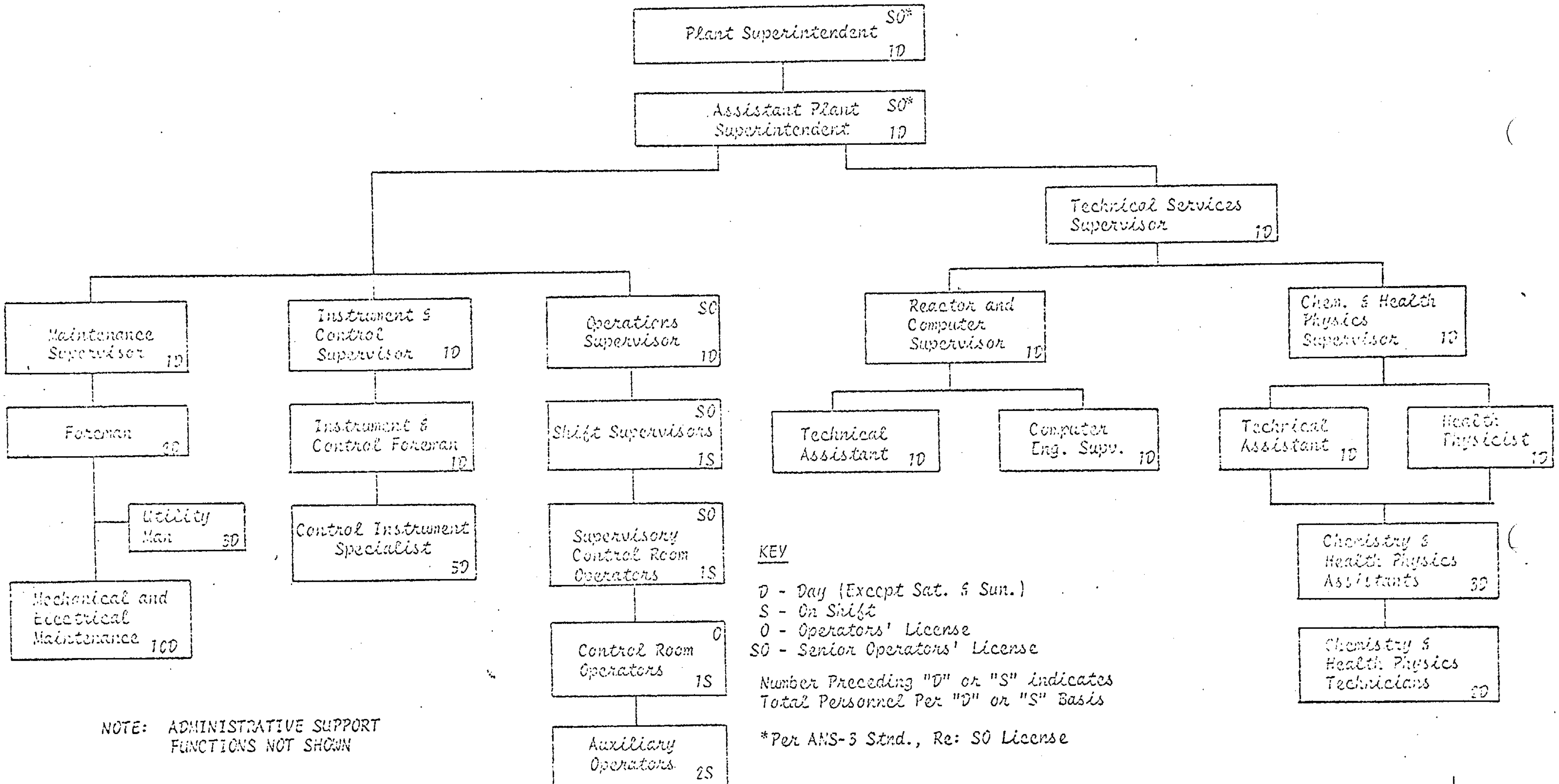


FIGURE 6.1.2 NORMAL FUNCTIONAL ORGANIZATIONAL CHART

## 6.2 REVIEW AND AUDIT

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

A. Plant Operations Review Committee

1. Membership

a. Chairman: Plant Superintendent

b. Vice-Chairman: Assistant Plant Superintendent

c. Vice-Chairman: Technical Services Supervisor 25

d. Vice-Chairman: Engineering Support Supervisor

e. Operations Supervisor

f. Maintenance Supervisor

g. Reactor and Computer Supervisor

h. Chemistry and Health Physics Supervisor

i. Instrument and Control Supervisor

j. Health Physicist

2. Qualifications

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

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

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

NOTE: For purposes of satisfying a quorum, Vice Chairmen may be considered members providing such Vice-Chairmen are not presiding over the meeting. 25

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 TO LICENSE NO. DPR-28

(CHANGE NO. 25 TO TECHNICAL SPECIFICATIONS)

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

## Introduction

By letter dated March 20, 1975, Vermont Yankee Nuclear Power Corporation (VYNPC) requested a change to the Technical Specifications appended to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed change involves corrections necessitated by previous license amendments, corrections of typographical and specification reference errors, clarification of intent and organizational changes which have occurred at the corporate and operational levels.

## Discussion

The following deletions from the Appendix A Technical Specifications have been made on the basis of previous amendments and changes which were approved but the corrections were overlooked:

1. Note 9 to Table 3.1.1 - The delay time of 300 milliseconds for reactor scram upon actuation of the turbine control valve fast closure signal was deleted by Amendment 12 dated December 3, 1974, thus select rod insert associated with this delay was removed. The sentence relating to this action has been deleted.

2. Tables 4.1.1 and 4.1.2, "Reactor Pressure-Permissive" - The removal of the reactor pressure permissive feature was approved by Amendment 9 dated October 23, 1974 thus the surveillance for this feature should have been removed. This function has been deleted from these tables.

3. Table 4.2.1, "LPCI Crosstie Monitor" - As stated in the Safety Evaluation for Amendment 11 dated December 3, 1974 which approved the LPCIS modification, a surveillance requirement was added for the LPCI crosstie valve monitor to verify that the valve is closed even though the power

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to the valve is disconnected. Verification that the valve is in the closed position requires an instrument check and not a functional test. The importance of the LPCIS valve being closed for proper LPCIS operation requires a daily instrument check as required for similar functions. The surveillance requirement has been changed from a functional test once a month to the instrument check once a day as was intended by the staff at the time the LPCI modification was approved by Amendment 11.

4. Table 4.2.1, "Auxiliary Power Monitor" - In the revision of page 51 to the table for Amendment 11 dated December 3, 1974, a typographical error was made by transposing the surveillance requirements associated with the calibration for a deleted trip function resulting in an inadvertent change in calibration frequency from "every refueling" to "every 3 months". This error has been corrected.

5. Specifications 3.6.G and 4.6.G, "Recirculation Pump Flow Mismatch" - Amendment 11 dated December 3, 1974 approved the LPCIS modification and deleted the LPCIS recirculation loop selection logic system. These specifications imposed requirements on the pump flow for the two recirculation loops associated with the selection logic system. When this logic system was removed, the need for information regarding pump flow mismatch was removed. These unnecessary specifications have been deleted.

Notes 1 and 2 to Table 3.2.4 have been revised to more clearly describe the off gas system isolation instrumentation.for clarification. The word changes do not alter the operation or function of the instrumentation but do more clearly define the action to be taken when the instrumentation becomes inoperable. The current notes imply redundant channels for all systems whereas only the air ejector off-gas radiation monitoring system has ever had two channels. The notes have been restated to clarify this matter.

In Specification 4.3.C.2, the word "during" has been added for clarification that the scram time measurements of control rod insertion can be made either during or following a controlled shutdown of the reactor. The change has been made to eliminate possible misinterpretation of when scram time measurements of control rods can be performed to meet the surveillance requirements.

Figures 6.1.1 and 6.1.2 have been changed to reflect the current organization at the corporate and operations level. Consistent with the increased importance placed on security matters and quality control independence, the corporate organization chart shows these groups to be reporting directly to the assistant vice president for operations. The other changes are minor

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and do not affect the operation of the facility. The only change to the operation organization is the addition of a technical services supervisor to direct all technical services for plant operation. The remaining operations groups continue to report directly to the plant superintendent. These changes should improve the effectiveness of the organization and therefore are acceptable.

Three minor changes have been made in the Plant Operations Review Committee structure. The first change is a change in title for one of the vice-chairman designees from "Technical Assistant to the Plant Superintendent" to "Technical Services Supervisor". The second change involves the addition of a third vice chairman, "the Engineering Support Supervisor". The third change is the addition of a note to clarify the intent that if a Vice-Chairman is not presiding and is present at a meeting he may be counted as a member to meet the quorum requirements. These changes do not alter the effectiveness of the Committee and therefore are acceptable.

All other changes made involve changes in the Bases for clarification and do not affect the specifications for operation of the facility. These changes include a reference change in Bases 3.6.B, corrections in Bases 3.6.F and deletion in Bases 4.7.A.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: MAY 21 1975

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 14 to Facility Operating License No. DPR-28 issued to Vermont Yankee Nuclear Power Corporation which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station, located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment permits changes to the organization for the corporate and operational functions, incorporates corrections necessitated by previous license amendments and makes changes to clarify the intent of the current scram time surveillance requirements and the existing off gas system instrumentation design.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

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For further details with respect to this action, see (1) the application for amendment dated March 20, 1975, (2) Amendment No. 14 to License No. DPR-28, with Change No. 25 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Brooks Memorial Library at 224 Main Street, Brattleboro, Vermont 05301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, ~~D. C.~~ 20555, Attention: ~~Director~~, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 26th day of May, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

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**SAFETY EVALUATION BY OFFICE OF NUCLEAR REACTOR REGULATION**

**SUPPORTING AMENDMENT NO. 14 TO LICENSE NO. DPR-28**

**(CHANGE NO. 25 TO TECHNICAL SPECIFICATIONS)**

**VERMONT YANKEE NUCLEAR POWER CORPORATION**

**VERMONT YANKEE NUCLEAR POWER STATION**

**DOCKET NO. 50-271**

## Introduction

By letter dated March 20, 1975, Vermont Yankee Nuclear Power Corporation (VYNPC) requested a change to the Technical Specifications appended to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed change involves corrections necessitated by previous license amendments, corrections of typographical and specification reference errors, clarification of [illegible] and organizational changes which have occurred at the corporate and operational levels.

## Discussion

The following deletions from the Appendix A Technical Specifications have been made on the basis of previous amendments and changes which were approved but the corrections were overlooked:

1. Note 9 to Table 3.1.1 - The delay time of 300 milliseconds for reactor scram upon actuation of the turbine control valve fast closure signal was deleted by Amendment 12 dated December 3, 1974, thus select rod insert associated with this delay was removed. The sentence relating to this action has been deleted.

2. Tables 4.1.1 and 4.1.2, "Reactor Pressure-Permissive" - The removal of the reactor pressure permissive feature was approved by Amendment 9 dated October 23, 1974 thus the surveillance for this feature should have been removed. This function has been deleted from these tables.

Move to page 2

3. 4 Notes 1 and 2 to Table 3.2.4 ~~These notes~~ have been revised to more clearly describe the off gas system isolation instrumentation. for clarification The word changes do not alter the operation or function of the instrumentation but do more clearly define the action to be taken when the instrumentation becomes inoperable. ~~The notes have been restated.~~ The current notes imply redundant channels for all systems whereas only the air ejector off-gas radiation monitoring system has two channels. The notes have been restated to ~~[illegible]~~ [illegible]

Clarify this matter.

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~~4.~~ 3. Table 4.2.1, "LPCI Crosstie Monitor" - As stated in the Safety Evaluation for Amendment 11 dated December 3, 1974 which approved the LPCIS modification, a surveillance requirement was added for the LPCI crosstie valve monitor to verify that the valve is closed even though the power to the valve is disconnected. Verification that the valve is in the closed position requires an instrument check and not a functional test. The importance of the LPCIS valve being closed for proper LPCIS operation requires a daily instrument check as required for similar functions. The ~~new~~ surveillance requirement has been ~~added~~. changed from a function test once a month to the ~~[illegible]~~ instrument check once a day as was intended by the staff at the time of the LPCI modification was approved by Amendment 11.

4. ~~5.~~ Table 4.2.1, "Auxiliary Power Monitor" - In the revision of page 51 to the table, a typographical error was made by transposing the surveillance requirements associated with the calibration for a deleted trip function resulting in an inadvertent change in calibration frequency from "every refueling" to "every 3 months". This error has been corrected.

for Amendment 11 dated [illegible] 3, 1974

~~6.~~ 5. Specifications 3.6.G and 4.6.G, "Recirculation Pump Flow Mismatch" - Amendment 11 dated December 3, 1974 approved the LPCIS modification and deleted the LPCIS recirculation loop selection logic system. These specifications imposed requirements on the pump flow for the two recirculation loops associated with the selection logic system. When this logic system was removed, the need for information regarding pump flow mismatch was removed. These unnecessary specifications have been deleted.

add [illegible] 3. att

In Specification 4.3.C.2, the word "during" has been added for clarification that the scram time measurements of control rod insertion ~~are~~ can be made either during or following a controlled shutdown of the reactor. The change has been made to eliminate possible misinterpretation of when scram time measurements of [illegible] can be performed to meet the surveillance requirements.

Figures 6.1.1 and 6.1.2 have been changed to reflect the current organization at the corporate and operations level. Consistent with the increased importance placed on security matters and quality control independence, the corporate organization chart shows these groups to be reporting directly to the assistant vice president for operations. The other changes are minor and do not affect the operation of the facility. The only change to the operation organization is the addition of a technical services supervisor to direct all technical services for plant operation. The remaining operations groups continue to report directly to the plant superintendent. These changes should improve the effectiveness of the organization and therefore are acceptable.

~~Two~~ Three minor changes have been made in the Plant Operations Review Committee structure. The first change is a change in title for one of the vice-chairman designees from "Technical Assistant to the Plant Superintendent"

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the addition of a "third vice-chairman"; the third change is the addition of the Engineering Support Supervisor".

to "Technical Services Supervisor". The second change involves a note to clarify the intent that if a Vice-Chairman is not presiding and is present at a meeting he may be counted as a member to meet the quorum requirements. These changes do not alter the effectiveness of the Committee and therefore are acceptable.

All other changes made involve changes in the Bases for clarification and do not affect the specifications for operation of the facility. These changes include a change in Bases 3.6.B, corrections in Bases 3.6.F and deletion in Bases 4.7.A.

## Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

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The amendment permits changes to the organization for the corporate and operational functions, ~~and~~ incorporates corrections necessitated by previous license amendments and makes changes to clarify the intent of the current scram time surveillance requirements and the existing off gas system instrumentation design.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

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