Docket No. 50-271

April 10/74

FDAnderson, L:ORB #2
RMDiggs, L:ORB #2
NDube, L:OPS
MJinks, DRA (4)
SKari, L:RP
SVarga, ~~XXRRX~~ L:RP
BScharf, DRA (15)
PCollins, L:OLB
VMoore, L:LWR-2

Distribution
Docket File
AEC PDR
Local PDR
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JRBuchanan, ORNL
TBAbernathy, DTIE
DJSkovholt, L:OR
TJCarter, L:OR
ACRS (16)
RO (3)
OGC
DLZiemann, L:ORB #2
RVoller, L:OA
Change No. 17
License No. DPR-28

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Albert A. Cree, President
77 Grove Street
Rutland, Vermont 05701

Gentlemen:

As discussed in our letters dated January 17 and January 28, 1974, authorizing Changes Nos. 13 and 15 to the Technical Specifications of Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station, we are completing the review and reissuance of the Technical Specifications with this change.

On the basis of our review reflected in the enclosed Safety Evaluation, we have concluded that the proposed changes do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility License No. DPR-28 are hereby changed as set forth in Attachment A to this letter.

As discussed between our respective staffs and stated in our related Safety Evaluation, meetings are being held to discuss revisions to the Appendix B Technical Specifications which relate to the environmental aspects of the Vermont Yankee plant operations. Some of these revisions will result in the transfer of the Radiological Monitoring Program (Specifications 3.9.D and 4.9.D) from Appendix A to Appendix B of the Technical Specifications appended to the Vermont Yankee Nuclear facility license.

Sincerely,

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Const. permit change auth.
CP 2

~~Enclosures and cc: See next page~~

| | | | | | |
|---|---|---|---|---|---|
| OFFICE | L:ORB #2 | L:ORB #2 | L:ORB #2 | L:OR | OGC |
| X7403 SURNAME | FDAnderson:sjh | RMDiggs | DLZiemann | DJSkovholt | |
| DATE | 2/5/74 | 2/4/74 | 2/11/74 | 2/ /74 | 2/10/74 |

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

Enclosures:
1. Attachment A - Changes to Technical Specifications
2. Safety Evaluation

cc w/enclosures:

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ATTACHMENT A

CHANGE NO. 17 TO THE TECHNICAL SPECIFICATIONS

FACILITY LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

1. Replace Appendix A cover page dated March 21, 1972, with the enclosed dated Appendix A cover page.

2. Replace Table of Contents, pages i through iii, with the enclosed Table of Contents, pages i through iv.

3. Replace Sections 3.8 and 4.8, "Station Radioactive Waste Control Systems", pages 141 through 153, with the enclosed Sections 3.8 and 4.8, "Radioactive Effluents", and Sections 3.9 and 4.9, "Radioactive Effluent Monitoring Systems", pages 147 through 172.

4. Replace Sections 3.9 and 4.9, "Station Electrical Power Systems", pages 154 through 161, with the enclosed Sections 3.10 and 4.10, "Auxiliary Electrical Power Systems", pages 173 through 180.

5. Replace Sections 3.10 and 4.10, "Station Refueling Systems", pages 162 through 167, with the enclosed Sections 3.11 and 4.11, "Refueling", pages 181 through 187.

6. Replace Section 5, "Design Features", pages 168 and 169, with the enclosed Section 5, "Design Features", pages 188 and 189.

7. Replace Section 6, "Administrative Controls", pages 170 through 195, with the enclosed Section 6, "Administrative Controls", pages 190 through 208.

APPENDIX A

TO

OPERATING LICENSE DPR-28

TECHNICAL SPECIFICATIONS

AND BASES

FOR

VERMONT YANKEE NUCLEAR POWER STATION

VERNON, VERMONT

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

Reissued by
Changes Nos. 13, 15, and 17
Dated 1/17/74, 1/28/74, and 4/10/74

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TABLE OF CONTENTS (CONT)

3.8 LIMITING CONDITIONS FOR OPERATION

## 3.8 RADIOACTIVE EFFLUENTS

### Applicability

Applies to the controlled release of all radioactive effluents from the plant.

### Objective

To assure that the release of radioactive material is kept as low as practicable and, in any event, is within the limits specified in 10 CFR Part 20. To meet the "as low as practicable" concept, the actual release of radioactive material should not represent more than a few percent of the 10 CFR Part 20 limits.

### Specifications

A. Liquid Effluents

1. The maximum concentration of radioactive material, except tritium and dissolved noble gases, at the point of discharge to the Connecticut River shall not exceed $1 \times 10^{-7}$ uCi/ml unless the discharge is controlled on a radionuclide basis in accordance with Appendix B, Table II, Column 2 of 10 CFR Part 20 and notes 1-5 thereto.

2. The maximum concentration of tritium at the point of discharge to the Connecticut River shall not exceed $3 \times 10^{-3}$ uCi/ml.

3. The maximum concentration of dissolved noble gases at the point of discharge to the Connecticut River shall not exceed $4 \times 10^{-5}$ uCi/ml.

4.8 SURVEILLANCE REQUIREMENT

## 4.8 RADIOACTIVE EFFLUENTS

### Applicability

Applies to the required surveillance during controlled release of all radioactive effluents from the plant.

### Objective

To ascertain that all radioactive effluents released from the plant are kept as low as practicable and, in any event, are within the limits specified in 10 CFR Part 20 and these Technical Specifications.

### Specifications

A. Liquid Effluents

1. Radioactive liquid effluents released from the facility shall be continuously monitored. The liquid effluent monitor shall be calibrated to measure the radioactivity released and shall have applicable alarm set points consistent with Specification 3.8.A.1 with appropriate operator action.

2. A sample of representative radioactive liquid effluents shall be analyzed for dissolved noble gases and tritium at least once per quarter to demonstrate that the concentration of dissolved noble gases or tritium is not greater than Specification 3.8.A.7.

3.8 LIMITING CONDITIONS FOR OPERATION

4. If the quantity of radioactive materials released during a calendar quarter, except tritium and dissolved noble gases, exceed 2.5 curies, the following actions shall be taken:

   a. Investigate to identify the causes for such release rates,

   b. Define and initiate a program to reduce such release rates to the as low as practicable levels, and

   c. Provide a report describing these actions within 30 days as an unusual event (see Specification 6.7).

5. If the average concentration of tritium exceeds 1 x $10^{-5}$ uCi/ml or the average concentration of dissolved noble gases exceeds 8 x $10^{-7}$ uCi/ml during a calendar quarter, the actions specified in Specification 3.8.A.4 shall be taken.

6. If the quantity of radioactive materials released during a calendar quarter, except tritium and dissolved noble gases, exceeds 10 curies, appropriate corrective action shall be initiated to reduce the quantity of radioactive materials to within the limit of Specification 3.8.A.4.

7. If the average concentration of radioactive materials during a calendar quarter exceeds 4 x $10^{-5}$ uCi/ml for tritium or 3 x $10^{-6}$ uCi/ml for dissolved noble gases at the point of discharge to the Connecticut River, appropriate corrective action shall be initiated to reduce concentrations of radioactive materials to within the limits of Specification 3.8.A.5.

4.8 SURVEILLANCE REQUIREMENT

3. Measurements shall be made on a representative sample of each batch of radioactive liquid effluents released and station records retained of the total (mCi) and concentration (uCi/ml) of gross radioactivity and volume (liters) and estimates made of the total quantity of water (liters) used to dilute the liquid effluent prior to release to the Connecticut River.

4. Each batch of radioactive liquid effluent released shall be analyzed for gross gamma radioactivity with efficiency of counting determined prior to any discharge but not more frequently than once per quarter from gamma scans. Isotopic analyses shall be used to demonstrate that the quantity and concentration of radioactivity are not greater than Specifications 3.8.A.1, 3.8.A.2, and 3.8.A.3.

5. A monthly proportional composite sample, comprising an aliquot of each batch released during a month, shall be analyzed by gamma spectroscopy. In addition, a portion of the composite sample shall be analyzed for tritium, SR-89, SR-90, gross beta and gross alpha radioactivity.

## 3.8 LIMITING CONDITIONS FOR OPERATION

8. The equipment installed in the liquid radioactive waste system shall be maintained and operated to process, as a minimum, all liquids prior to their discharge when the radioactivity, exclusive of tritium and dissolved noble gases, released during a calendar quarter exceeds 1.25 curies.

9. If the limits of Specifications 3.8.A.1. through 3. cannot be met, radioactive liquid effluents shall not be released.

B. Radioactive Liquid Storage

The maximum gross radioactivity in liquid storage in the Waste Sample Tanks, Floor Drain Sample Tanks, and the Waste Surge Tank shall be less than 3.2 curies except for tritium and dissolved noble gases. If this condition cannot be met, the liquids in these tanks shall be recycled to tanks within the radwaste facility until the condition is met.

C. Gaseous Effluents

1. Gross Radioactivity

a. The maximum release rate of gross radioactivity from the plant shall not exceed: $0.08/\bar{E}_\gamma$ Ci/sec where $\bar{E}_\gamma$ is the average gamma decay energy from the gaseous effluents in Mev/disintegration.

## 4.8 SURVEILLANCE REQUIREMENT

6. Two independent samples from a liquid waste tank shall be taken and analyzed for gross gamma radioactivity and the valve line-up checked prior to discharge of liquid effluents from that tank.

7. The results of independent samples analyses and valve performance checks shall be logged.

B. Radioactive Liquid Storage

1. A sample shall be taken, analyzed, and recorded within 72 hours of each addition to a liquid waste storage tank to which Specification 3.8.B. applies.

2. If the sample analysis indicates that the total radioactivity in liquid storage in the Waste Sample Tanks, Floor Drain Sample Tanks, and the Waste Surge Tank exceeds 3.2 curies, except for tritium and dissolved noble gases, the liquids in these tanks shall be recycled to redu the radioactivity to less than 3.2 curies within 24 hours of the sampling.

C. Gaseous Effluents

1. Gross Radioactivity

a. Radioactive gases released from the plant stack shall be continuously monitored. The stack gas monitor shall be calibrated to measure the radioactivity released and shall have applicable alarm set points consistent with Specifications 3.8.C.1.a and 3.8.C.1.b.

## 3.8 LIMITING CONDITIONS FOR OPERATION

b. If the release rate of gross radioactivity from the plant stack averaged over a calendar quarter exceeds 16 percent of the limit in Specification 3.8.C.1.a, appropriate corrective action shall be initiated to reduce releases of gross radioactivity to within the limits in Specification 3.8.C.1.c.

c. If the release rate of gross radioactivity from the plant stack averaged over a calendar quarter exceeds four percent of the limits in Specification 3.8.C.1.a, the actions in Specification 3.8.A. 4 shall be taken.

d. If the maximum release rate limits of Specification 3.8.C.1.a are not met following a routine surveillance check, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

## 4.8 SURVEILLANCE REQUIREMENT

b. Measurements of gross radioactivity shall be made continuously and plant records retained of the quantities of radioactive gases released. During the period of release, hourly measurements of meteorological parameters shall be recorded. Records of isotopic analysis shall also be maintained.

c. A steam jet air ejector off-gas sample shall be taken and an isotopic analysis of at least six fission product gases; Xe-138, Xe-135, Xe-133, Kr-88, Kr-85m, Kr-87 shall be made at least weekly and following each refueling or other occurrence which could alter significantly the mixture of radionuclides.

d. Samples of off-gas effluents shall be taken at least every 96 hours and gamma radioactivity determined by gross or isotopic analysis.

e. Gaseous release of tritium shall be calculated on a quarterly basis.

f. Radioactive gases released from the air ejector shall be continuously monitored. The air ejector monitor shall be calibrated to measure radioactivity released and shall have a high alarm set point equivalent to 0.3 Ci/sec after 30 minutes decay as given in Specification 3.2.D, Table 3.2.4.

Operator action shall be to determine if system isolation is required for high radioactivity alarm.

g. Radioactive gases entering the charcoal bed system shall be continuously monitored. The pre-charcoal bed monitor shall be calibrated to measure radioactivity in the delay line. Operator action shall be to determine if system isolation is required for low radioactivity levels.

h. Radioactive gases released from the charcoal bed system shall be continuously monitored prior to entering the delay line to the reactor stack. The post-charcoal bed monitor shall be calibrated to measure radioactivity leaving the charcoal bed and shall have an automatic isolation set point equivalent to Specification 3.8.C.1.a. Operator action shall be to determine if system isolation is required for low radioactivity levels.

2. Radioiodine

a. The maximum release rate of radioiodine 131 from the plant shall not exceed 0.48 uCi/sec.

2. Radioiodine

a. Radioiodine released from the plant stack shall be continuously sampled. The stack charcoal cartridge shall be removed and counted weekly when the

3.8 LIMITING CONDITIONS FOR OPERATION

b. If the release rate of I-131 averaged over a calendar quarter exceeds 4 percent of the limit in Specification 3.8.C.2.a., appropriate corrective action shall be initiated to reduce releases of I-131 to within the limits in Specification 3.8.C.2.c.

c. If the release rate of I-131 averaged over a calendar quarter exceeds 2 percent of the limit in Specification 3.8.C.2.a, the actions in Specification 3.8.A.4 shall be taken.

d. If the maximum release rate limits of Specification 3.8.C.2.a are not met following a routine surveillance check, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

3. Radioactive Particulates

a. The maximum release rate of radioactive particulates with half lives greater than 8 days from the plant shall not exceed $1.6 \times 10^3$ $\overline{MPC}_a$ Ci/sec where $\overline{MPC}_a$ is the composite maximum permissible concentration in air as determined in Appendix B, Table II, Column 1 of 10 CFR Part 20 and Notes 1-5 thereto.

4.8 SURVEILLANCE REQUIREMENT

measured release rate of I-131 is less than Specification 3.8.C.2.b; otherwise, the stack charcoal cartridge shall be removed and counted daily.

b. Station records of all I-131 released from the reactor stack shall be maintained on the basis of all stack charcoal cartridges counted.

c. A determination shall be made of the total I-131 released weekly. An analysis shall be performed on a sample at least monthly for I-133 and I-135.

3. Radioactive Particulates

a. Radioactive material in particulate form released from the plant stack shall be continuously sampled. The stack particulate filter shall be removed and counted weekly when the measured release rate of radioactive particulates with half lives greater than 8 days is less than Specification 3.8.C.3.b; otherwise the stack particulate filter shall be removed and counted daily.

3.8 LIMITING CONDITIONS FOR OPERATION

b. If the release rate of radioactive particulates with half lives greater than 8 days averaged over a calendar quarter exceeds 8 percent of the limit in Specification 3.8.C.3.a., appropriate corrective action shall be initiated to reduce releases of radioactive particulates to within the limits in Specification 3.8.C.3.c.

c. If the release rate of radioactive particulates with half lives greater than 8 days averaged over a calendar quarter exceeds 2 percent of the limit in Specification 3.8.C.3.a, the actions in Specification 3.8.A.4 shall be taken.

d. If the maximum release rate limits of Specification 3.8.C.3.a are not met following a routine surveillance check, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown condition within 24 hours.

4.8 SURVEILLANCE REQUIREMENTS

b. Station records of all radioactive material in particulate form with half lives greater than 8 days released from the reactor stack shall be maintained on the basis of all stack particulate filters counted.

c. A determination shall be made of the total radioactive material in particulate form with half lives greater than 8 days released weekly. The particulate filters shall be removed and analyzed at least weekly for gross beta-particulate radioactivity with half lives greater than 8 days. Monthly, a composite of those filters used during the month shall be prepared and analyzed for the principal gamma emitting radionuclides.

d. Analysis for Sr-89, Sr-90, and gross alpha radioactivity shall be made quarterly.

Bases:

## 3.8 RADIOACTIVE EFFLUENTS

### A. Liquid Effluents

The radioactive liquid effluents from the Vermont Yankee plant will be controlled on a batch basis with each batch being processed by such method or methods appropriate for the quantity of material determined to be present. Those batches in which the radioactivity concentrations are sufficiently low to allow release to the discharge canal are diluted with dilution pump flow (and condenser circulating water flow except when plant is operating on closed cycle) in order to achieve the allowable concentrations. The radioactive liquid will be sampled and analyzed for gross radioactivity prior to release to the discharge canal, thus providing a means of obtaining information on effluents to be released so that appropriate release rates will be established.

Liquid effluent release rate will be controlled in terms of the quantity released and/or the concentration in the discharge canal. In the case of unidentified mixtures, such concentration limit is based on the assumption that the entire content is made up of the most restrictive isotope. Such a limit assures that even if a person obtained all of his daily water intake from such a source, the resultant dose would not exceed that specified in 10 CFR 20. If radioactive effluents are released to unrestricted areas on a radionuclide basis, the MPC shall be determined and controlled in the cooling water discharge canal in accordance with Appendix B, Table II, Column 2 of 10 CFR 20 and Notes 1-5 thereto. Each batch to be released will conform to 10 CFR 20 release limits on an instantaneous basis. The maximum concentration limit for dissolved noble gases was derived on the basis that Xe-135 was the controlling radioisotope whose MPC for air (submersion) was converted to an equivalent concentration in water using the ICRP method for conversion.

In order to limit liquid effluent releases to as low as practicable quantities, quarterly release quantities have been established which would require investigative actions at 2.5 curies and restricted operation at 10 curies. These release levels are significantly below 10 CFR 20 limits and are factors of 2 and 8, respectively, above the as low as practicable objective of 5 curies per year release level. The same relationship has been maintained for tritium and dissolved noble gas released. Liquid wastes discharged to the river need not be processed if the quarterly release rate is less than 1.25 curies and the equipment is not operable.

3.8 (cont'd)

B. Radioactive Liquid Waste Storage

The maximum gross radioactivity in liquid storage in the specified tanks has been limited on the basis of an accidental spill from all stated tanks due to a seismic event great enough to damage them. Assuming a low river flow of 108 $ft^3/sec$, a day period over which the radioactive liquid wastes are diluted in the river, and consumption of the water by individuals at standard man consumption rate (3000 ml/day), the single intake by an individual would not exceed one-third the yearly intake allowable by 10 CFR 20 for unidentified radioisotopes ($1 \times 10^{-7}$ uCi/ml). The factor of 3 was applied to 10 CFR 20 limits as recommended for situations in which population groups could be exposed.

C. Gaseous Effluents

1. Gross Radioactivity

Detailed dose calculations for several locations offsite have been made and are described in Appendix E of the FSAR. These calculations consider site meteorology, buoyancy characteristics, and isotopic content of the effluent. Independent dose calculations for several locations off site have been made by the AEC staff, and the most critical one was chosen to set the maximum release rate. This point is 370 meters to the south at the site boundary. The method utilized onsite meteorological data developed by the licensee and utilized diffusion assumptions appropriate to the site.

The method utilized by the staff is described in Section 7-5.2.5 of "Meteorology and Atomic Energy - 1968", equation 7.63 being used. The results of these calculations are conservative and thus chosen to be used as the basis of establishment of the limits. Based on these calculations, a continuous release rate of gross radioactivity in the amount of $0.08/\bar{E}_\gamma$ curies/sec from the plant stack would not result in offsite annual whole body doses in excess of the limits specified in 10 CFR 20 of 500 mRem. The $\bar{E}_\gamma$ determination need consider only the average gamma energy per disintegration since the controlling whole body dose is due to the cloud passage over the receptor and not cloud submersion in which the beta dose could be additive.

3.8.C (cont'd)

In order to limit gross radioactivity releases in gaseous effluents to as low as practicable, quarterly average release rates have been established which would require investigative actions at 4 percent of the maximum release rate and restricted operation at 16 percent of the maximum release rate. These release rates are significantly below 10 CFR 20 limits and are factors of 2 and 8, respectively, above the as low as practicable objective of 2 percent of 10 CFR Part 20 limits.

2. Radioiodine

Detailed dose calculations for several locations off site have been made by the AEC staff and the most critical 22.5° sectors were determined to be at 3200 m to the northwest and at 2400 m to the west-northwest of the site. The method utilized by the staff determines the annual average concentration in the most critical sector which could result in an annual thyroid dose of 1500 mRem to a child that drank the milk from a cow continuously consuming the grass from the critical sector. The maximum annual average diffusion parameter (X/Q) from the licensee's meteorological data for the elevated release is $9 \times 10^{-7}$ sec/m$^3$. Based on these calculations, a continuous release rate of I-131 in the amount of 0.48 uCi/sec from the plant stack would not result in offsite annual thyroid doses in excess of the limits specified in 10 CFR 20 of 1500 mRem. The nearest worst real cow approach has not been utilized in this analysis as permitted by Regulatory Guide 1.42 but may be applied if actual release rates from the plant stack indicate the need.

In order to limit radioiodine releases in gaseous releases to as low as practicable, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and restricted operation at 4 percent of the maximum release rate. These release rates are significantly below 10 CFR 20 limits and are factors of 2 and 4, respectively, above the as low as practicable objectives of 1 percent of 10 CFR Part 20 limits.

3. Radioactive Particulates

The AEC staff performed an analysis similar to that used to determine the maximum release rate for I-131 for the radioactive particulates. A reduction factor of 700 on the $\overline{MPC}_a$ to allow for possible ecological chain effects similar to those associated with radioiodine was used. The annual average diffusion parameter value of $9 \times 10^{-7}$ sec/m$^3$ was used as determined for the most critical sector for an elevated release. Based on these calculations, a continuous release rate of radioactive particulates with half lives greater than 8 days in the amount of $1.6 \times 10^3$ $\overline{MPC}_a$ Ci/sec from the plant stack would not result in offsite annual organ doses in excess of the limits specified in 10 CFR 20. The resultant organ doses are not additive to those caused by the radioiodine or gross radioactivity releases.

3.8.C (cont'd)

In order to limit radioactive particulate releases in gaseous effluents to as low as practicable, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and restricted operation at 8 percent of the maximum release rate. These release rates are significantly below 10 CFR 20 limits and are factors of 2 and 8, respectively, above the as low as practicable objectives of 1 percent of 10 CFR Part 20 limits.

The specifications allow for a routine surveillance check by the operator prior to initiating an orderly shutdown of the reactor if the maximum radioactivity release rate limits are exceeded. The routine surveillance check by the operator is to be used to verify the results of the monitoring method which has indicated a violation of the limits and to determine that this condition is continuing and does not represent either an instrument error or failure or a system transient which cannot be immediately corrected by the operator. In most cases the off-gas system will have been isolated automatically by the signal from the off-gas monitoring system to prevent additional releases of radioactivity. However, operation of the reactor will not be affected by this system isolation for a reasonable time period during which the operator may take corrective action.

Bases:

4.8 RADIOACTIVE EFFLUENTS

A. Liquid Effluents

Surveillance of liquid effluent releases is required to assure compliance with the limits established in Specification 3.8.A. Experience has shown that continuous monitoring with appropriate alarm and operator action is necessary to prevent inadvertent liquid releases with high radioactive concentrations. Quarterly sampling is adequate to assure compliance with limits established for tritium and dissolved noble gas releases in liquid effluents from light-water reactors. Representative samples of liquid effluents are taken from each batch of liquid effluents released to identify the type and quantity of radioisotopes present in the releases. These results are used to determine total radioactive liquid effluent releases from the plant. The sensitivity of measurement should be related to the objective for meeting the as low as practicable requirement of 10 CFR Part 20. The accuracy of measurement becomes more critical for those releases that approach the concentration limits specified by 10 CFR Part 20.

B. Radioactive Liquid Storage

The sampling frequency has been established so that if the maximum amount of gross radioactivity is exceeded, action can be taken to reduce the radioactivity to a level below the specified limit. For the quantities of radioactivity considered limiting for liquid storage, the accuracy of measurement should be appropriately low and the measurements quite accurate.

C. Gaseous Effluents

Measurements of the gross radioactivity from the plant stack must be continuously monitored for possible changes in the release rates from the augmented off-gas system. Additional measurements are made continuously at the steam jet air ejector to evaluate the core condition and the quantity of radioactivity being added to the modified off-gas system. The measurements obtained by sampling and isotopic analysis define the releases to the environs. Quarterly analysis for tritium is adequate to define such releases to the environs. The sensitivity of measurement should be related to the objective for meeting the as low as practicable requirements of 10 CFR Part 20. The accuracy of measurement becomes more critical for those releases that approach the release rate equivalent to 10 CFR Part 20 limits.

4.8.C (cont'd)

The average gamma energy per disintegration used in the equation of Specification 3.8.C.1 will be based on the average composition of gases determined by the latest isotopic analyses. Considering the above, Specification 3.8.C.1 gives equations to be used in the airborne effluents from the plant stack which will assure that offsite doses are not in excess of the limits specified in 10 CFR 20 and will be as low as practicable. The gamma energy per disintegration for those radioisotopes determined to be present from the isotopic analyses shall be as given in "Table of Isotopes", C. M. Lederer, J. M. Hollander, and I. Perlman, Sixth Edition, 1967. For Kr-89 and Xe-138, the gamma energy per disintegration shall be as given in "Energy Release from the Decay of Fission Products", Nuclear Science and Engineering: 3,726-746 (1958) until values are published in "Table of Isotopes". Using these reference gamma energies per disintegration with the composition of radiogases in the plant stack releases, the average gamma energy per disintegration, $\bar{E}_\gamma$, shall be determined.

Isotopic analysis will be performed on samples taken from the steam jet air ejector. These samples will be used in an isotopic analysis for Xe-138, Xe-135, Xe-133, Kr-88, Kr-87, and Kr-85m, which is calculated to be approximately 90 percent of the noble gas emission entering the augmented off-gas system. The remaining noble gases will be calculated from empirical ratios with the measured gases. Such calculations will be made for the various gases down to a release rate of 100 uCi/sec. Argon 41 will not be measured routinely since it cannot be measured in the presence of the other noble gases.

Measurements of the gross radioactivity between the steam jet air ejector and the reactor stack must be continuously monitored for possible failures of the various subsystems in series (recombiner, delay line, charcoal bed, delay line) as well as changes in the effectiveness of these subsystems to reduce the effluent releases. The pre-charcoal bed monitor is a backup to the steam jet air ejector monitor; a low reading will indicate to the operator possible recombiner or line failure unless the input source term to the recombiner has also been reduced. The automatic isolation set point for high radiation after the charcoal bed indicates a break down of the augmented off-gas system (probably charcoal bed) and is a backup system to the reactor stack monitor. A low reading for the post-charcoal bed monitor will indicate to the operator possible charcoal bed tank failure unless the input source term to the charcoal bed system was reduced for a period of time several days previously.

4.8.C (cont'd)

The release of radioiodine from the reactor stack is monitored by the use of charcoal cartridges which integrate the releases over the sampling period of one to seven days. Frequency of removal is dependent upon the release level measured on the previously removed charcoal cartridge. The analysis performed for I-133 and I-135 indicates the contribution of these radioiodines to the possible inhalation doses. The sensitivity of measurement and accuracy of measurement are most critical for the radioiodine releases since such releases are expected to be the controlling releases for meeting the as low as practicable objective.

The release of radioactive particulates with half lives greater than eight days from the reactor stack is monitored by the use of particulate filters which integrate the releases over the sampling period of one to seven days. All other aspects of particulate release measurements are similar to those discussed for radioiodine release measurements. The sensitivity of measurements and accuracy of measurements for particulates are equivalent to those necessary for gross radioactivity and radioiodines.

## 3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

### Applicability

Applies to the radioactive effluent monitoring system which performs a surveillance, protective, or controlling function on the release of radioactive effluents from the plant.

### Objective

To assure the operability of the radioactive effluent monitoring system.

### Specifications

A. Liquid Effluent Monitoring System

1. Either the liquid effluent monitor on the discharge line or the discharge canal sampler shall be operable at all times during the course of a radioactive liquid discharge.

2. If Specification 3.9.A.1 cannot be met, discharge of radioactive liquid effluents shall be discontinued until grab samples in the discharge structure can be taken and analyzed or the monitor made operable. If grab samples are taken, such samples shall be taken hourly and analyzed during the course of the discharge.

## 4.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

### Applicability

Applies to the required surveillance of the radioactive effluent monitoring system.

### Objective

To specify the type and frequency of surveillance to be applied to the radioactive effluent monitoring systems.

### Specifications

A. Liquid Effluent Monitoring System

1. The liquid effluent monitor on the discharge line shall be calibrated quarterly, functionally tested by injecting a simulated electrical signal into the measurement channels monthly, and have an instrument check at least daily during discharges. The operability of the sampler shall be verified on a daily basis during discharges.

2. At least each operating cycle, initiation of the alarm system for the liquid effluent monitor and operation of the isolation valve on the discharge line shall be performed.

## 3.9 LIMITING CONDITIONS FOR OPERATION

### B. GASEOUS EFFLUENT MONITORING SYSTEM

#### 1. Plant Stack

a. At least one of the two plant stack monitoring systems, including the charcoal cartridge and particulate filter, shall be operable at all times.

b. From and after the date that both plant stack monitoring systems are found inoperable, continued reactor operation is permissible for a period up to 24 hours provided that the reactor is operating at steady state conditions, both the reactor building vent duct monitors and the monitor after the charcoal bed systems are operating to satisfy Specification 3.8.C.1.a release limits, and continuous air monitors are operating to satisfy Specifications 3.8.C.2.a and 3.8.C.3.a release limits.

c. From and after the date that both plant stack monitoring systems are found inoperable, continued reactor operation is permissible for a period up to 7 days provided the reactor is operating at steady state conditions, both the reactor building vent duct monitor and the monitor after the charcoal bed system are operating to satisfy Specification 3.8.C.1.b release limits, and continuous air monitors are operating to satisfy Specifications 3.8.C.2.b and 3.8.C.3.b release limits.

## 4.9 SURVEILLANCE REQUIREMENTS

### B. GASEOUS EFFLUENT MONITORING SYSTEM

#### 1. Plant Stack

a. The plant stack radiation monitors shall be functionally tested by injecting a simulated electrical signal into the measurement channels monthly and calibrated quarterly with an appropriate radiation source Each monitor, as described, shall have an instrument check at least daily.

b. The measurement of I-131 and particulates with half lives greater than 8 days obtained from the continuous air monitor samples shall be used to determine quantities of radioiodines and radioactive particulates released during periods when both stack charcoal cartridges and/or particulate filters are inoperable.

## 3.9 LIMITING CONDITIONS FOR OPERATION

d. If the above Specification 3.9.B.1 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours.

2. Augmented Off-Gas System

a. If the hydrogen concentration in the off-gas downstream of the recombiners reaches four percent, the off-gas flow shall be stopped automatically by closing the valves upstream and downstream of the recombiners.

b. Except as specified in Specification 3.9.B.2.c below, at least one hydrogen monitor downstream of the recombiners shall be operating during power operation.

c. If the above specified required hydrogen monitors are not available, an orderly transfer of the off-gas effluents from the operating recombiner to the standby recombiner shall be made.

d. At least one of the two radiation monitoring systems between the recombiner and the first charcoal bed shall be operable during operation of the augmented off-gas system.

e. If Specification 3.9.B.2.d cannot be met, continued operation of the augmented off-gas system is permissible for a period up to 7 days provided the reactor is operating at steady state conditions and a steam jet air ejector off-gas

## 4.9 SURVEILLANCE REQUIREMENTS

2. Augmented Off-Gas System

a. The hydrogen monitors shall be functionally tested by injecting a simulated electrical signal into the measurement channels monthly and calibrated quarterly with an appropriate gas mixture source. Each monitor shall have an instrument check at least daily.

b. The augmented off-gas radiation monitors shall be functionally tested by injecting a simulated electrical signal into the measurement channels monthly and calibrated quarterly with an appropriate radiation source. Each monitor, as described, shall have an instrument check at least daily.

c. The off-gas system pressure and temperature shall be used to determine system failures during periods when the radiation monitoring systems between the recombiner and the charcoal bed system are inoperable or the radiation monitoring system between the charcoal bed system and the reactor stack is inoperable. Operator action as specified in Specification 4.8.C.1 shall be performed for off-normal pressure levels.

3.9 LIMITING CONDITIONS FOR OPERATION

radiation monitor is operating and the off-gas system pressure and temperature are measured continuously.

f. 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.

g. If Specification 3.9.B.2.f cannot be met, continued operation of the augmented off-gas system is permissible for a period up to 7 days provided at least one of the two plant stack monitoring systems is operating and the off-gas system pressure and temperature are measured continuously.

C. Liquid Process Monitors

When the systems are in operation, the activity in the service water and the cooling tower water systems shall be continually monitored and either of the following conditions shall be met:

1. The process liquid radiation monitor shall be operable, or

2. Samples shall be taken daily and analyzed for gross radioactivity.

4.9 SURVEILLANCE REQUIREMENTS

C. Liquid Process Monitors

The liquid process monitors shall be functionally tested by injecting a simulated electrical signal into the measurement channels monthly and calibrated quarterly with an appropriate radiation source. Each monitor shall have an instrument check at least daily.

3.9 LIMITING CONDITIONS FOR OPERATION

D. Radiological Monitoring Program

1. A radiological monitoring program shall be conducted to monitor the effects of plant operation on the environment.

2. Frequency of sampling shall be in accordance with Table 3.9.1.

4.9 SURVEILLANCE REQUIREMENTS

D. Radiological Monitoring Program

1. A radiological monitoring program as given in Table 3.9.1 shall be conducted.

2. Type of analysis and media monitored shall be in accordance with Table 3.9.1.

TABLE 3.9.1

ATMOSPHERIC AND TERRESTRIAL MONITORING STATIONS

| Site | Location & Elevation | Distance Miles | Site Quadrant | Sample Media |
|---|---|---|---|---|
| ZONE-I | | | | |
| AT1.0 | East Bank of Connecticut River | 0.8 | NE | A-D-(c) V |
| AT1.1 | No. Hinsdale, N.H. (280') | 2.5 | NNW | A-D-(c) V |
| AT1.2 | Hinsdale, N.H. (260') | 1.9 | E | A-D-(c) V |
| AT1.3 | River Station #3.3, Vt. (450') | 1.2 | SSE | A-D-(c) V |
| AT1.4 | Central Park Sta., Vt. (450') | 0.9 | SSW | A-D-(c) V |
| AT1.5 | Guilford Rd., Vernon (580') | 1.5 | W | A-D-(c) V |
| AT1.6-1.16 | Site Boundary, Vernon (300') | Variable | | D (11 locations) |
| AT1.17 | Vernon School | --- | SW | D |
| ZONE-II | | | | |
| AT2.1 | Hogback Mt., Vt. (1670') | 15.5 | WNW | A-D-(c) V |
| AT2.2 | Spofford Lake, N.H. (750') | 10.0 | NNE | A-D-(c) V |
| AT2.3 | Northfield, Mass. (350') | 7.0 | SSE | A-D-(c) V |

| Symbols | Media | Planned Collection Frequency | Type of Analysis |
|---|---|---|---|
| A | Air Particulate, Continuous Sampling | Weekly<br>Quarterly | Gross Beta and, per Note 3, Gross Alpha for weekly. Isotopic analysis for those particulate isotopes and the significant particulate daughters of all isotopes shown to be present in the release for quarterly. |
| D | Direct Radiation, TLD Dosimeters | Monthly | Gamma Dose |
| V | Vegetation (Mixed Grass) | Note (4)<br>Quarterly | Gross Beta, Gamma Scan |
| (c) | Charcoal Filter Cartridges (during plant operation) | Weekly | Gamma Scan (I-131) or equivalent |

TABLE 3.9.1 (CONT'D)

## GROUND WATER STATIONS

| Site | Designation | Location | Distance Miles | Direction | Collection Frequency | Type of Analysis |
|---|---|---|---|---|---|---|
| **ZONE I** | | | | | | |
| GW1.1 | VY Plant (well) | On-Site | 0.0 | --- | Quarterly | Gross Alpha per Note 3, Gross Beta Gamma Scan |
| GW1.2 | NH #1 (well) | Private Home | 0.5 | ENE | Quarterly | " |
| GW1.3 | Vt. #17 (spring) | Hunt House | 0.2 | SW | Quarterly | " |
| GW1.4 | Vt. #27 (spring) | Vernon Dam | 0.5 | SSW | Quarterly | " |
| GW1.5 | Vt. X (well) | Vernon Nursing | 1.1 | SSE | Quarterly | " |
| GW1.6 | Vt. #32 (well) | Private Home | 1.1 | SE | Quarterly | " |
| **ZONE II** | | | | | | |
| GW2.1 | Supply wells | Brattleboro Country Club | 7.0 | NNW | Quarterly | " |

## MILK MONITORING STATIONS

| Site | Designation | Location | Distance Miles | Direction | Collection Frequency | Type of Analysis |
|---|---|---|---|---|---|---|
| **ZONE I** | | | | | | |
| M1.1 | | Farm - Raw Vernon | 0.5 | NW | Monthly* | Gross Beta, Gamma Isotopic, and except when cattle are fed on stored food I-131. |
| M1.2 | | Farm - Raw Vernon | 1.8 | S | Monthly* | |
| M1.3 | | Farm - Raw Hinsdale | 2.0 | E | Monthly* | |
| **ZONE II** | | | | | | |
| M2.1 | | Brattleboro - Processed | 10. | NNW | Monthly* | |

*A sample will be analyzed quarterly for Sr-90.

VYNPS
TABLE 3.9.1 (CONT'D)

RIVER MONITORING STATIONS

| Site | Location | River - Miles From Discharge | Sample Media |
| --- | --- | --- | --- |
| ZONE I | | | |
| R1.1 | Schell Bridge E. Northfield (Mass) | 7.3 Downstream | W-S-B |
| R1.2 | B&M R.R. Bridge-State Line (Vt) | 5.6 Downstream | W-S-B |
| R1.3 | Power Cable Crossing - South of Vernon Dam (Vt) | 1.5 Downstream | W-S-B |
| R1.4 | Vernon Pond at Discharge Structure (Vt) | --- | Note (1) W-S-B |
| R1.5 | River Station #149 Gravel Pit (Vt) | 3.2 Upstream | W-S-B |
| ZONE II | | | |
| R2.1 | Rte. 9 Highway Bridge (N.H.) | 7.7 Upstream | W-S-B |

AQUATIC VEGETATION STATIONS

| Site | Location | River - Miles From Discharge | Sample Media |
| --- | --- | --- | --- |
| S1.1 | Swamp Area | ∿0.3 Downstream | AP |
| S1.2 | Swamp Area | ∿0.3 Upstream | AP |

TABLE 3.9.1 (CONT'D)

| Symbols | Media | Planned Collection Frequency | Type of Analysis |
|---|---|---|---|
| W | Water (Grab) | Monthly | Gross Beta and, per Note 3, Gross Alpha |
| | | Quarterly | Tritium and Gamma Spectrum |
| S | River Sediment | Quarterly Note (2) | Gross Beta, Gross Alpha per Note 3, Gamma Spectrum |
| B | Biological Specimen (Fish, Plankton, Benthic Organisms, Water Fowl and, as appropriate, Deer, Rabbit, etc. | Quarterly Note (2) | Gross Beta, Gross Alpha per Note 3, Gamma Spectrum |
| AP | Aquatic Plants | Quarterly Note (2) | Gross Beta, Gamma Spectrum |

Note (1) Composite monthly water samples will be collected during operational monitoring period.

(2) Approximate collection frequency dependent upon the availability of aquatic organisms and limitations imposed by severe winter river conditions.

(3) Gross alpha analyses will be made if alpha emitting isotopes are shown to be present in the releases.

(4) Approximate collection frequency dependent upon availability of vegetation samples and limitations imposed by severe winter conditions.

Bases:

## 3.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

### A. Liquid Effluent Monitoring System

Liquid effluent monitoring has been provided as a backup to the sampling and analysis performed prior to liquid discharge from the liquid waste tanks. If these systems become inoperable, grab samples can be taken from the discharge structure and analyzed for the concentration resulting from any discharges.

### B. Gaseous Effluent Monitoring System

The plant stack monitoring system is provided to measure the radioactive effluents released to the environs from the stack. The results are used as a basis for evaluating the environmental impact of these releases and for correlation with the environmental monitoring program. If these systems become inoperable for limited periods of time, the reactor building vent duct and the augmented off-gas monitoring systems can be used as backup systems for the gross radioactivity measurements and continuous air monitors can be used as a backup system to the charcoal cartridges and particulate filters.

The hydrogen monitors are used to detect possible hydrogen buildups which could result in a possible hydrogen explosion. Isolation of the off-gas flow would prevent the hydrogen explosion and possible damage to the augmented off-gas system.

The augmented off-gas monitoring systems are provided to measure the radioactive effluents from the steam jet air ejector through the recombiner to the delay line and the charcoal beds through the second delay line to the reactor stack. The results are used to determine the effectiveness of the off-gas system to reduce the effluent releases and are used by the operator to minimize radioactive releases in case of system failure. If these systems become inoperable for limited periods of time, the steam jet air ejector monitors and reactor stack monitors can be used in conjunction with the off-gas system pressure and temperature monitors to detect system failure and radioactive releases.

### C. Liquid Process Monitors

The liquid process monitoring system is provided to detect any failures in the service water or cooling tower water systems which could result in radioactivity from the primary coolant system being released indirectly to the environs through these systems. If these systems become inoperable, samples can be taken and analyzed daily without reducing the effectiveness of the protection to detect system failure and prevent significant releases of radioactivity to the environs.

3.9 (cont'd)

D. Radiological Monitoring Program

A two-zone sample collection network has been established for radiological surveillance. Samples are collected in Zone I at locations in the vicinity of the station where concentrations of station effluents may be detectable. These samples are compared to samples which have been collected simultaneously at locations in Zone II where the concentration of station effluents is expected to be negligible. The Zone II samples provide a running background which will make it possible to distinguish significant radioactivity introduced into the environment by the operation of the station from that introduced by nuclear detonations and other sources.

Aquatic biological monitoring stations have been established at river sampling points upstream and downstream from the plant discharge structure. The function of monitoring the aquatic environment is to identify and determine the magnitude of any radionuclide reconcentration in the aquatic food chain.

Bases:

## 4.9 RADIOACTIVE EFFLUENT MONITORING SYSTEMS

### A. Liquid Effluent Monitoring System

Experience has shown that daily checking of the radiation monitor and sampler is required with monthly testing and quarterly calibration to assure proper operation of the equipment. Performance of the isolation function and initiation of the alarm system during each operating cycle is adequate to assure proper operation if necessary.

### B. Gaseous Effluent Monitoring System

Experience has shown that a daily check with monthly testing and quarterly calibration assures proper operation of the monitoring equipment both for the hydrogen monitors and the radiation monitors.

### C. Liquid Process Monitors

Experience has shown that a daily check with monthly testing and quarterly calibration assures proper operation of the monitoring equipment.

### D. Radiological Monitoring Program

Special attention is given to gamma spectrum analysis of media in order to identify and reference nuclides present prior to plant startup. As the environmental monitoring changes from the preoperational stage, alpha, beta, and gamma spectra of sample media is observed for changes in types of activity present. The program is designed to be modified easily to include those isotopes which analysis shows to be present in the effluents.

The program is responsive to surveillance recommendations made by the U. S. Fish and Wildlife Service in their correspondence with AEC.

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 Auxiliary Electrical Power Systems

Applicability

Applies to the auxiliary electrical power systems.

Objective

To assure an adequate supply of electrical power for operation of those systems required for reactor safety.

Specification

A. Normal Operation

The reactor shall not be made critical unless all of the following conditions are satisfied.

1. Diesel Generators

Both emergency diesel generators shall be operable and capable of starting and reaching rated voltage and frequency in not more than 13 seconds.

4.10 SURVEILLANCE REQUIREMENTS

4.10 Auxiliary Electrical Power Systems

Applicability

Applies to the periodic testing requirements of the auxiliary electrical power systems.

Objective

To verify the operability of the auxiliary electrical power systems.

Specification

A. Normal Operation

1. Diesel Generators

a. Each diesel generator shall be started and loaded once a month to demonstrate operational readiness. The test shall continue until the diesel engine and the generator are at equilibrium temperature at expected maximum emergency loading not to exceed the continuous rating. During this test, the diesel starting time to reach rated voltage and frequency shall be logged, the air compressor shall be checked for operation and its ability to recharge air receivers, and the diesel fuel oil transfer pumps shall be operated.

2. Batteries

All station 24 and 125 volt battery systems and the switchyard 125 volt battery system shall be operable. The associated battery chargers for the 24 volt batteries, two of the three battery chargers for the 125 volt station battery chargers, and one of the two 125 volt switchyard battery chargers, shall be operable.

b. The undervoltage automatic starting circuit of each diesel generator shall be tested once a month.

c. Once per operating cycle, the actual conditions under which the diesel generators are required to start automatically will be simulated and a test conducted to demonstrate that they will start within 13 seconds and accept the emergency load and start each load within the specified starting time. The results shall be logged.

2. Batteries

a. Every week the specific gravity and voltage of the pilot cell and temperature of adjacent cells and overall battery voltage shall be measured and logged.

b. Every three months the voltage of each cell to nearest 0.01 volt and specific gravity of each cell to the nearest 0.005 sp.gr. shall be measured and logged.

c. Once each operating cycle each station 125 volt battery shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be measured after the discharge test and logged.

3. Emergency Buses

The emergency 4160 volt buses 3 and 4, and 480 volt buses 8, 9 and 89 shall be energized and operable.

4. Off-site Power

a. At least one offsite transmission line and the start-up transformer in service.

b. One of the following additional sources of delayed access power:

The main step-up transformer and unit auxiliary transformer available and capable of supplying power to the emergency 4160 volt buses or,

The 4160 volt tie line to Vernon Hydroelectric Station capable of supplying power to either of the two emergency 4160 volt buses.

3. Emergency Buses

The emergency 4160 volt buses and 480 volt buses shall be checked daily.

4. Off-site Power

The status of the off-site power sources shall be checked daily.

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 3.9.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.

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.

3.10 LIMITING CONDITIONS FOR OPERATION

3. Offsite Power

a. From and after the date that the start-up transformer and one diesel generator or associated buses are made or found to be inoperable for any reason, reactor operation may continue provided the requirements of Specification 3.5.H.1 are satisfied.

b. From and after the date that both delayed-access offsite power sources become unavailable, reactor operation may continue for seven days provided both emergency diesel generators, associated buses and all low pressure core and containment cooling systems are operable.

C. Diesel Fuel

There shall be a minimum of 25,000 usable gallons of diesel fuel in the diesel fuel oil storage tank.

4.10 SURVEILLANCE REQUIREMENTS

3. Offsite Power

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

b. When it is determined that both delayed-access offsite power sources are unavailable, both diesel-generators, associated buses and all low pressure core and containment cooling systems shall be demonstrated to be operable immediately and daily thereafter.

C. Diesel Fuel

1. The quantity of diesel generator fuel shall be logged weekly and after each operation of the unit.

2. Once a month a sample of diesel fuel shall be taken and checked for quality. The quality shall be within the applicable limits specified on Table 1 of ASTM D975-68 and logged.

Bases:

## 3.10 AUXILIARY ELECTRICAL POWER SYSTEMS

A. The objective of this specification is to assure that adequate power will be available to operate the emergency safeguards equipment. Adequate power can be provided by any one of the following sources: The start-up transformer, backfeed through the main transformer, the 4160 volt line from the Vernon Hydro-electric Station or either of the two diesel generators. The backfeed through the main transformer and the 4160 volt Vernon line are both delayed-access offsite power sources. Backfeeding through the main transformer can be accomplished by disconnecting the main generator from the main transformer and energizing the auxiliary transformer from the 345 kv switchyard through the main transformer. The time necessary to perform this disconnection is approximately six hours. The 4160 volt line from the Vernon Hydroelectric Station can be connected to either of the two emergency buses within seconds by simple manual switching operation in the main control room.

This Specification assures that at least two offsite and two onsite power sources will be available before the reactor is taken beyond "just critical" testing. In addition to assuring power source availability, all of the associated switchgear must be operable as specified to assure that the emergency cooling equipment can be operated, if required, from the power sources.

Station service power is supplied to the station through either the unit auxiliary transformer or the start-up transformer. In order to start-up the station, the start-up transformer is required to supply the station auxiliary load. After the unit is synchronized to the system, the unit auxiliary transformer carries the station auxiliary load, except for the station cooling tower loads which are always supplied by the start-up transformer. The station cooling tower loads are not required to perform an engineered safety feature function in the event of an accident therefore an alternate source of power is not essential.

A battery charger is supplied with each of the two 125 volt d-c main station batteries. In addition, a spare charger is available and can supply power to either battery system. Since this alternative source is available, one battery charger can be allowed out of service for maintenance and repairs.

B. Adequate power is available to operate the emergency safeguards equipment from the start-up transformer or for minimum engineered safety features from either of the emergency diesel generators. Therefore, reactor operation is permitted for up to seven days with both delayed-access offsite power sources lost.

Each of the diesel generator units is capable of supplying 100 percent of the minimum emergency loads required under postulated design basis accident conditions. Each unit is physically and electrically independent of the other and of any offsite power source. Therefore, one diesel generator can be allowed out of service for a period of seven days without jeopardizing the safety of the station.

3.10.B (cont'd)

In the event that the start-up transformer is lost, adequate power is available to operate the emergency safeguards equipment from either of the emergency diesel generators or from either of the delayed-access offsite power sources. Also, in the event that both emergency diesel generators are lost, adequate power is available immediately to operate the emergency safeguards equipment from the start-up transformer or from either of the delayed access offsite power sources within six hours. The plant is designed to accept one hundred percent load rejection without adverse effects to the plant or the transmission system. Network stability analysis studies indicate that the loss of Vermont Yankee unit will not cause instability and consequent tripping of the emanating 345 kV and 115 kV lines. The Vernon feed is an independent source. Thus, the availability of the delayed-access offsite power sources is assured in the event of a turbine trip. Therefore, reactor operation is permitted with the start-up transformer out of service and with one diesel generator out of service provided the AEC is notified immediately of the event and restoration plans.

Either of the two station batteries has enough capacity to energize the vital buses and supply d-c power to the other emergency equipment for 8 hours without being recharged. Due to the high reliability of battery systems, one of the two batteries may be out of service for up to three days. This minimizes the probability of unwarranted shutdown by providing adequate time for reasonable repairs. A station battery is considered inoperable if more than one cell is out of service. A cell will be considered out of service if its float voltage is below 2.13 volts and the specific gravity is below 1.190 at 77°F.

The battery room is ventilated to prevent accumulation of hydrogen gas. With a complete loss of the ventilation system, the accumulation of hydrogen would not exceed 4 percent concentration in 16 days. Therefore, on loss of battery room ventilation, the use of portable ventilation equipment and daily sampling provide assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

C. The minimum diesel fuel supply of 25,000 gallons will supply one diesel generator for a minimum of seven days of operation satisfying the load requirements for the operation of the safeguards equipment. Additional fuel can be obtained and delivered to the site from nearby sources within the seven day period.

4.10 AUXILIARY ELECTRICAL POWER SYSTEMS

Bases:

A. The monthly tests of the diesel generators are conducted to check for equipment failures and deterioration. The test of the undervoltage automatic starting circuits will prove that each diesel will receive a start signal if a loss of voltage should occur on its emergency bus. The loading of each diesel generator is conducted to demonstrate proper operation at less than the continuous rating and at equilibrium operating conditions. Generator experience at other generator stations indicates that the testing frequency is adequat to assure a high reliability of operation should the system be required.

4.10.A (cont'd)

Both diesel generators have air compressors and air receivers tanks for starting. It is expected that the air compressors will run only infrequently. During the monthly check of the units each receiver will be drawn down below the point at which the compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

Following the tests of the units and at least weekly, the fuel volume remaining will be checked. At the end of the montly loads test of the diesel generator, the fuel oil transfer pump will be operated to refill the day tank and to check the operation of this pump. The day tank level indicator and alarm switches will be checked at this time.

The test of the diesels during each refueling interval will be more comprehensive in that it will functionally test the system; e.e., it will check starting and closure of breakers and sequencing of loads. The units will be started by simulation of a loss of coolant accident. In addition, a loss of normal power condition will be imposed to simulate a loss of offsite power. The timing sequence will be checked to assure proper loading in the time required. Periodic tests between refueling intervals check the capability of the units to start in the required time and to deliver the expected emergency load requirements. Periodic testing of the various components plus a functional test at a refueling interval are sufficient to maintain adequate reliability.

B. Although the station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The rated load discharge test provides adequate indication and assurance that the batteries have the specified ampere hour capacity. The rate of discharge during this test shall be in accordance with the manufacturer's discharge characteristic curves for the 125 volt station batteries. The results of these tests will be logged and compared with the manufacturers recommendations of acceptability.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-68. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-68.

3.11 LIMITING CONDITION FOR OPERATION

## 3.11 REFUELING

**Applicability:**

Applies to fuel handling and core reactivity limitations.

**Objective:**

To assure core reactivity is within capability of the control rods and to prevent criticality during refueling.

**Specification:**

A. Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks, listed below, shall be operable except as specified in Specifications 3.11.D and 3.11.E.

1. Control Rod Blocks

   a. Mode switch in Startup/Hot Standby and refueling platform over the reactor.

4.11 SURVEILLANCE REQUIREMENT

## 4.11 REFUELING

**Applicability:**

Applies to the periodic testing of those interlocks and instruments used during refueling.

**Objective:**

To verify the operability of instrumentation and interlocks used in refueling.

**Specification:**

A. Refueling Interlocks

Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

3.11 LIMITING CONDITION FOR OPERATION

b. Fuel on any refueling hoist and refueling platform over the reactor.

c. Mode switch in Refuel with one control rod withdrawal permit.

2. Refueling Platform Reverse Motion (toward reactor vessel) Block

a. Mode switch in Startup/Hot Stand-by.

b. Any control rod out and fuel on any refueling hoist.

3. Refueling Platform Hoists Blocks

a. Any control rod out and fuel on any refueling hoist over the vessel.

b. Hoist overload.

c. High position limitation.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being

4.11 SURVEILLANCE REQUIREMENT

B. Core Monitoring

Prior to making any alterations to the core the SRM's shall be functionally tested and checked for neutron response.

3.11 LIMITING CONDITION FOR OPERATION

moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special movable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected into the proper circuitry which contain the required rod blocks).

2. The SRM shall have a minimum of 3 cps with all rods fully inserted in the core.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of at least 36 feet.

4.11 SURVEILLANCE REQUIREMENT

Thereafter, the SRM's shall be checked daily for response.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool level shall be recorded daily.

## 3.11 LIMITING CONDITIONS FOR OPERATION

### D. Control Rod and Control Rod Drive Maintenance

A maximum of two non-adjacent control rods separated by more than two control cells in any direction, may be withdrawn from the core for the purpose of performing control rod and/or control rod drive maintenance provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed for one of the control rods on which maintenance is being performed. All other refueling interlocks shall be operable.

2. Specification 3.3.A.1 shall be met, or the control rod directional control valves for a minimum of eight control rods surrounding each drive out of service for maintenance shall be disarmed electrically and sufficient margin to criticality demonstrated.

3. SRMs shall be operable (a) in each core quadrant containing a control rod on which maintenance is being performed, and (b) in a quadrant adjacent to one of the quadrants specified in Specification 3.11.D.3.(a) above. Requirements for an SRM to be considered operable are given in Specification 3.11.B.

### E. Extended Core Maintenance

More than two control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

## 4.11 SURVEILLANCE REQUIREMENTS

### D. Control Rod and Control Rod Drive Maintenance

1. Sufficient control rods shall be withdrawn prior to performing this maintenance to demonstrate with a margin of 0.25 percent $\Delta k$ that the core can be made subcritical at any time during the maintenance with the strongest operable control rod fully withdrawn and all other operable rods fully inserted.

2. Alternately, if a minimum of eight control rods surrounding each control rod out of service for maintenance are to be fully inserted and have their directional control valves electrically disarmed, the 0.25 percent $\Delta k$ margin shall be met with the strongest control rod remaining in service during the maintenance period fully withdrawn.

### E. Extended Core Maintenance

Prior to control rod withdrawal for extended core maintenance, that control rod's control cell shall be verified to contain no fuel assemblies.

## 3.11 LIMITING CONDITION FOR OPERATION

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

2. SRMs shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in Specification 3.11.B,

F. Fuel Movement

Fuel shall not be moved or handled in the reactor core for 24 hours following reactor shutdown to cold shutdown conditions.

## 4.11 SURVEILLANCE REQUIREMENT

1. This surveillance requirement is the same as that given in Specification 4.11.A.

2. This surveillance requirement is the same as that given in Specification 4.11.B.

F. Fuel Movement

Prior to any fuel handling or movement in the reactor core, the licensed operator shall verify that the reactor has been in the cold shutdown condition for a minimum of 24 hours.

Bases:

3.11 & 4.11 REFUELING

A. During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.2 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

The addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist.

Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

B. The SRMs are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRMs in or adjacent to any core quadrant where fuel or control rods are being moved assured adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored.

C. To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 36 feet is established because it would be a significant change from the normal level, well above a level to assure adequate cooling (just above active fuel).

3.11 & 4.11 (cont'd)

D. During certain periods, it is desirable to perform maintenance on two control rods and/or control rod drives at the same time. This specification provides assurance that inadvertent criticality does not occur during such maintenance.

The maintenance is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling operations as explained in Part A of these Bases. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated with the control rods remaining in service insures that inadvertent criticality cannot occur during this maintenance. The shutdown margin is verified by demonstrating that the core is shut down even if the strongest control rod remaining in service is fully withdrawn. Disarming the directional control valves does not inhibit control rod scram capability.

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as inservice inspection requirements, examination of the core support plate, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in the Bases for Specification 3.11.A. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

F. The intent of this specification is to assure that the reactor core has been in the cold shutdown condition for at least 24 hours following power operation and prior to fuel handling or movement. The safety analysis for the postulated refueling accident assumed that the reactor had been shutdown for 24 hours for fission product decay prior to any fuel handling which could result in dropping of a fuel assembly.

## 5.0 DESIGN FEATURES

### 5.1 Site

The station is located on the property on the west bank of the Connecticut River in the Town of Vernon, Vermont, which the Vermont Yankee Nuclear Power Corporation either owns or to which it has perpetual rights and easements. The plan of the site is shown on Figure 2.2-4 of the FSAR. The minimum distance to the boundary of the exclusion area as defined in 10 CRF 100.3 is 910 feet.

No part of the site shall be sold or leased and no structure shall be located on the site except structures owned by the Vermont Yankee Nuclear Power Corporation or related utility companies and used in conjunction with normal utility operations.

### 5.2 Reactor

A. The core shall consist of not more than 368 fuel assemblies of 49 fuel rods each.

B. The reactor core shall contain 89 cruciform-shaped control rods. The control material shall be boron carbide power ($B_4C$).

### 5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.2-3 of the FSAR. The applicable design codes shall be as described in subsection 4.2 of the FSAR.

### 5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the FSAR.

B. The secondary containment shall be as described in subsection 5.3 of the FSAR and the applicable codes shall be as described in Section 12.0 of the FSAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in subsection 5.2 of the FSAR.

5.5 Spent and New Fuel Storage

A. The new fuel storage facility shall be such that the effective multiplication factor ($k_{eff}$) of the fuel, dry is less than 0.90 and flooded is less than 0.95.

B. The $K_{eff}$ of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

## 6.0 ADMINISTRATIVE CONTROLS

Administrative control 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 Superintendent or the Assistant Plant Superintendent has on-site responsiblility for the safe and efficient operation of the facility.

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 these Technical Specifications, the Plant Superintendent 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 (See Table 6.1.1).

1. A licensed Senior Operator shall be present on site at all times when there is fuel in the reactor.

2. Licensed Operators on site shall be in accordance with Table 6.1.1, one of which must be in the control room at all times when fuel is in the reactor.

3. A licensed Senior Operator shall be in charge of any refueling operation.

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 Superintendent
   b. Assistant Plant Superintendent
   c. Chemistry and Health Physics Supervisor
   d. Operations Supervisor
   e. Reactor and Computer Supervisor
   f. Maintenance Supervisor
   g. Instrument and Control Supervisor
   h. Shift Supervisors

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

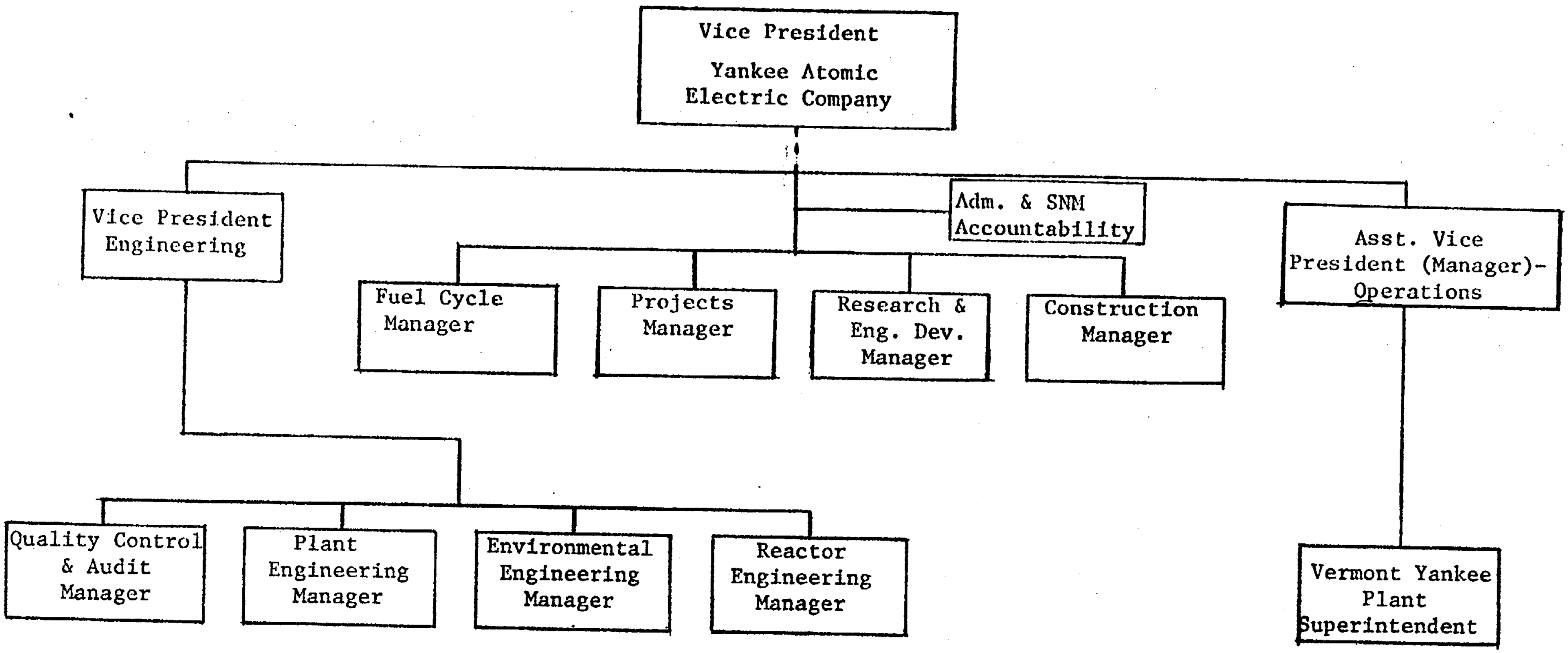
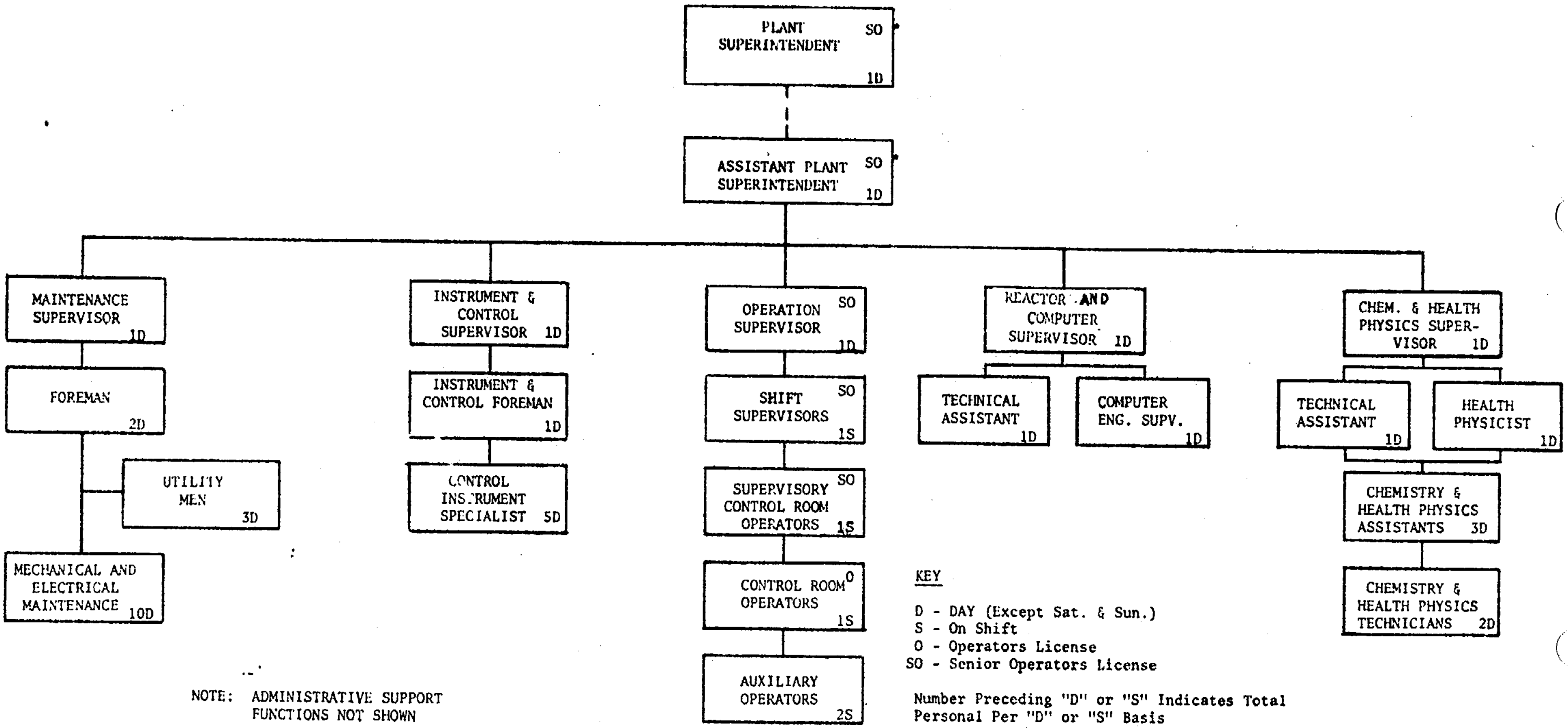



Figure 6.1.1 CORPORATE ORGANIZATION

VERMONT YANKEE NUCLEAR POWER CORPORATION




NOTE: ADMINISTRATIVE SUPPORT FUNCTIONS NOT SHOWN

KEY

D - DAY (Except Sat. & Sun.)
S - On Shift
O - Operators License
SO - Senior Operators License

Number Preceding "D" or "S" Indicates Total Personal Per "D" or "S" Basis

*Per ANS-3 Stnd., Re: SO License

FIGURE 6.1.2 NORMAL FUNCTIONAL ORGANIZATION CHART

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 |
|---|---|
| Operations Supervisor | Senior Operator |
| Shift Supervisor | Senior Operator |
| Supervisory Control Room Operator | Operator |
| Control Room Operator | Operator |

| MINIMUM SHIFT CREW PERSONNEL & LICENSE REQUIREMENTS | Conditions | | |
|---|---|---|---|
| | Normal Operation | Plant Startup | Cold Shutdown |
| Shift Supervisor | (1) | (1) | - |
| Supervisory Control Room Operator | (1) | (1) | (1) |
| Control Room Operator | (1) | (1) | (1) |
| Auxiliary Operator | (2) | (2) | (1) |
| Senior Operators License | (1) | (1) | (1) |
| Operators License | (2) | (2) | (1) |

## 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 Operation Review Committee

1. Membership

a. Chairman: Plant Superintendent

b. Vice Chairman: Assistant Plant Superintendent

c. Vice Chairman: Technical Assistant to the Plant Superintendent

d. Operations Supervisor

e. Maintenance Supervisor

f. Reactor and Conputer Supervisor

g. Chemistry and Health Physics Supervisor

h. Instrument and Control Supervisor

i. 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.

5. Designated alternates shall be from other plant personnel in the appropriate disciplines or from the Technical Services Group; however, there shall be no more than three (3) alternate members serving on the committee at any one time.

6. Responsibilities

   a. Review proposed normal, abnormal and emergency operating procedures. Review all proposed maintenance procedures and proposed changes to those procedures; and any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

   b. Review proposed tests and experiments.

   c. Review proposed changes to Technical Specifications.

   d. Review proposed changes or modifications to plant systems or equipment, which changes would require a change in procedures in (a) above.

   e. Review plant operations to detect any potential safety hazards.

   f. Investigate reported instances of violations of Technical Specifications, such investigations to include reporting, evaluation and recommendations to prevent recurrence, to the Manager of Operations in the Westboro Office of Yankee Atomic Electric Company (See Figure 6.2.1).

   g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Nuclear Safety Audit and Review Committee.

7. Authority

   a. The Plant Operation Review Committee shall be advisory.

   b. The Plant Operation Review Committee shall recommend to the Plant Superintendent approval or disapproval of proposals under Items f (a) through (d) above.

      1. In the event of disagreement between the recommendations of the Plant Operation Review Committee and the actions contemplated by the Plant Superintendent, the course determined by the Plant Superintendent to be the more conservative will be followed with immediate notification to the Manager of Operations.

c. The Plant Operation Review Committee shall make tentative determinations as to whether or not proposals considered by the Committee involves unreviewed safety questions. This determination shall be subject to review by the Nuclear Safety Audit and Review Committee.

8. Records

Minutes shall be kept at the plant of all meetings of the Plant Operation Review Committee and copies shall be sent to the Manager of Operations and the Nuclear Safety Audit and Review Committee.

B. Nuclear Safety Audit and Review Committee

1. The Committee shall consist of at least six (6) persons:

a. Chairman

b. Vice Chairman

c. Four technically qualified persons who are not members of the plant staff.

d. No more than three members shall be selected from the organization reporting to the Manager of Operations.

e. The Committee will obtain advice and counsel from scientific or technical personnel employed by the Company or other organizations whenever the Committee considers it necessary to obtain further scientific or technical assistance in carrying out its responsibilities.

f. The Committee membership and its Chairman and Vice Chairman shall be appointed by the Vermont Yankee Engineering Vice President or such person as he shall designate.

2. Qualifications

The Committee shall consist of a minimum of six (6) members or designated alternatives who as a group employ expertise in the following areas:

a. Nuclear Power Plant Technology

b. Reactor Operations

c. Utility Operations

d. Power Plant Design

e. Reactor Engineering

f. Radiation Safety

g. Safety Analysis

h. Instrumentation and Control

i. Metallurgy

3. Meeting Frequency: Semi-annually and as required on call of the Chairman.

4. Quorum: Chairman or Vice Chairman plus four members or designated alternatives.

5. Responsibilities

a. Review proposed changes to the operating license including Technical Specifications.

b. Review minutes of meetings of the Plant Operation Review Committee to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

c. Review matters including proposed changes or modifications to plant systems or equipment having safety significance or referred to it by the Plant Operation Review Committee or by the Plant Superintendent.

d. Conduct periodic audits of plant operations.

e. Investigate all reported instances of violations of Technical Specifications, reporting findings and recommendations to prevent recurrence to the Manager of Operations.

f. Perform special reviews and investigations and render reports thereon as requested by the Manager of Operations.

g. Review proposed tests and experiments and results thereof when applicable.

h. Review abnormal performance of plant equipment and anomalies.

i. Review unusual occurrences and incident which are reportable under the provisions of 10 CFR Part 20 and 10 CFR Part 50.

j. Review of occurrences if Safety Limits are exceeded.

6. Authority

a. Review proposed changes to the operating license including Technical Specifications and revised bases for submittal to the AEC.

b. Review proposed changes or modifications to plant systems or equipment, provided that such changes or modifications do not involve unreviewed safety questions.

c. Recommend to the Manager of Operations appropriate action to prevent recurrence of any violations of Technical Specifications.

d. Evaluate actions taken by the Plant Operation Review Committee.

7. Records

Minutes of all meetings of this committee shall be recorded. Copies of the minutes shall be forwarded to the Manager of Operations, the Engineering Vice President, the Plant Superintendent and any others that the Chairman may designate.

## 6.3 ACTION TO BE TAKEN IN THE EVENT OF AN ABNORMAL OCCURRENCE IN PLANT OPERATION

Applies to administrative action to be followed in the event of an abnormal occurrence in Plant operation.

Any abnormal occurrence shall be reported to the Manager of Operations and shall be reviewed by the Plant Operations Review Committee. This Committee shall prepare a separate, sequentially numbered, report for each abnormal occurrence. Each report shall describe the circumstances leading up to and resulting from the occurrence, the corrective action taken by the shift, an attempt to define the cause of the occurrence, and shall recommend appropriate action to prevent or reduce the probability of a repetition of the occurrence.

Copies of all such reports shall be submitted to the Chairman of the Nuclear Safety Audit and Review Committee for review and to the Manager of Operations for review and approval of any recommendations. All abnormal occurrences shall be reported in the Plant Operations Report.

## 6.4 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Applies to administrative action to be followed in the event a safety limit is exceeded.

If a safety limit is exceeded, the reactor shall be shutdown immediately. An immediate report shall be made to the Manager of Operations. A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations by the Plant Operations Review Committee shall also be prepared. This report shall be submitted to the Manager of Operations and the Chairman of the Nuclear Safety Audit and Review Committee.

Reactor operation shall not be resumed until authorized by the U. S. Atomic Energy Commission.

## 6.5 PLANT OPERATING PROCEDURES

A. Detailed written procedures involving nuclear safety, including applicable check-off lists and instructions, covering areas listed below shall be prepared and approved.

All procedures shall be adhered to.

1. Normal startup, operation and shutdown of systems and components of the facility.

2. Refueling operations.

3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected primary system leaks and abnormal reactivity changes.

4. Emergency conditions involving potential or actual release of radioactivity.

5. Preventive and corrective maintenance operations which could have an effect on the safety of the reactor.

6. Surveillance and testing requirements.

B. Radiation control standards and procedures shall be prepared, approved and maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR Part 20. This radiation protection program shall be organized to meet the requirements of 10 CFR Part 20.

1. Pursuant to 10 CFR 20.103(c)(1) and (3), allowance can be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this plant in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

   a. The limits provided in Section 20.103(a) and (b) are not exceeded.

   b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column 1, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, concentration value specified is based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in §20.101. These materials shall be subjected to applicable process and other engineering controls.

2. In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls in impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in Specification 6.5.B.1 are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment does not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.5.1, appended to this specification for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or phychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88, 2-1969). Such a program shall include:

(1) Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

(2) Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

(3) Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for operability immediately prior to use.

(4) Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

(5) Written operational and administrative procedures for proper use of respiratory protective equipment, including provisions for planned limitations on working times as necessitated by operational conditions.

(6) Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U. S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table I below. Equipment not approved under U. S. Bureau of Mines Approval Schedules may be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U. S. Bureau of Mines approved equipment of the same type, as specified in Table I below.

f. Unless otherwise authorized by the Commission, the licensee does not assign protection factors in excess of those specified in Table I below in selecting and using respiratory protective equipment.

3. These specifications with respect to the provisions of Section 20.103 shall be superseded by adoption of proposed changes to 10 CFR 20, Section 20.103, which would make this specification unnecessary.

4. In lieu of the "control device" of alarm signal required by paragraph 20.203(c)(2) of 10 CFR 20, 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 Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

The above procedure shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mRem/hr, except that locked doors shall be provided to prevent unauthorized entry into these areas and the keys to these locked doors shall be maintained under the administrative control of the Chemistry and Health Physics Supervisor and the Shift Supervisor on duty.

TABLE 6.5.1

PROTECTION FACTORS FOR RESPIRATORS

| DESCRIPTION | MODES 1/ | PROTECTION FACTORS 2/ PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE 3/ | GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES APPROVAL SCHEDULES* FOR EQUIPMENT CAPABLE OF PROVIDING AT LEAST EQUIVALENT PROTECTION FACTORS *or schedule superseding for equipment of type listed |
|---|---|---|---|
| I. AIR-PURIFYING RESPIRATORS | | | |
| Facepiece, half-mask 4/ 7/ | NP | 5 | 21B 30 CFR 14.4(b)(4) |
| Facepiece, full 7/ | NP | 100 | 21B 30 CFR 14.4(b)(5); 14F 30 CFR 13 |
| II. ATMOSPHERE-SUPPLYING RESPIRATOR | | | |
| 1. Airline Respirator | | | |
| Facepiece, half-mask | CF | 100 | 19B 30 CFR 12.2(c)(2) Type C(i) |
| Facepiece, full | CF | 1,000 | 19B 30 CFR 12.2(c)(2) Type C(i) |
| Facepiece, full 7/ | D | 100 | 19B 30 CFR 12.2(c)(2) Type C(ii) |
| Facepiece, full | PD | 1,000 | 19B 30 CFR 12.2(c)(2) Type C(iii) |
| Hood | CF | 5/ | 6/ |
| Suit | CF | 5/ | 6/ |
| 2. Self-contained breathing apparatus (SCBA) | | | |
| Facepiece, full 7/ | D | 100 | 13E 30 CFR 11.4(b)(2)(i) |
| Facepiece, full | PD | 1,000 | 13E 30 CFR 11.4(b)(2)(ii) |
| Facepiece, full | R | 1,000 | 13E 30 CFR 11.4(b)(1) |
| III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator | | Protection factor for type and mode of operation as listed above | 19B CFR 12.2(e) or applicable schedules as listed above |

1/, 2/, 3/, 4/, 5/, 6/, 7/, (These notes are on the following pages)

1/ See the following symbols:

CF: continuous flow
D : demand
NP: negative pressure (i.e., negative phase during inhalation)
PD: pressure demand (i.e., always positive pressure)
R : recirculating (closed circuit)

2/ (a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the face-piece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the folloiwng formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency (above 99.9% removal efficiency by U.S. Bureau of Mines type dioctyl phthalate (DOP) test) particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

3/ Excluding radioactive contaminants that present an absorption of submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote 5/, below concerning supplied-air suits and hoods.

4/ Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

5/ Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

6/ No approval schedules currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

7/ Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this table. The protection factors in this table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Bureau of Mines in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table 1 of this part are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

C. Procedures prepared for A and B above shall be reviewed and approved by the Plant Superintendent or the Assistant Plant Superintendent and the Manager of Operations.

D. Temporary changes to procedures described in Specification 6.5.A above which do not change the intent of the original procedure may be made with the concurrence of two individuals holding senior operator licenses. Such changes shall be documented and subsequently reviewed by the PORC and approved by the Plant Superintendent or Assistant Plant Superintendent.

E. Temporary changes to procedures described in Specification 6.5.B above may be made with the concurrence of an individual holding a senior operator license and the health physicist on duty.

F. Practice of site evacuation exercises shall be conducted annually, following emergency procedures and including a check of communications with off-site support groups. Annual reviews of the Emergency Plan shall be performed.

## 6.6 PLANT OPERATING RECORDS

A. Records and/or logs relative to the following items shall be kept in a manner convenient for review and shall be retained for at least five years:

1. Records of normal plant operation, including power levels and periods of operation at each power level.

2. Records of principal maintenance activities, including inspection and repair of principal items of equipment pertaining to nuclear safety.

3. Records of abnormal occurrences.

4. Records of periodic checks, inspection and/or calibrations performed to verify that surveillance requirements are being met.

5. Records of any special reactor test or experiments.

6. Records of changes made in the Operating Procedures.

7. Records of radioactive shipments.

B. Records and/or logs relative to the following items shall be recorded in a manner convenient for review and shall be retained for the life of the plant:

1. Records of substitution or replacement of principal items of equipment pertaining to nuclear safety.

2. Records of changes and drawing changes made to the plant as it is described in the Safety Analysis Report.

3. Records of plant radiation and contamination surveys.

4. Records of new and spent fuel inventory, transfers of fuel, and assembly histories.

5. Records of radioactivity in liquid and gaseous wastes released to the environment.

6. Records of radiation exposure for all plant personnel, including all contractors and visitors to the plant in accordance with 10 CFR 20.

7. Records of transient or operational cycling for those plant components that have been designed to operate safely for a limited number of transients or operational cycles.

8. Records of inservice inspections of the reactor coolant system.

9. Minutes of meetings of the Plant Operation Review Committee and the Nuclear Safety Audit and Review Board.

## 6.7 PLANT REPORTING REQUIREMENTS

The reporting information to be submitted to the USAEC in addition to the reports required by Title 10, Code of Federal Regulations shall be in accordance with the Regulatory Positions of Regulatory Guide 1.16, "Reporting of Operating Information", Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants", and Regulatory Guide 4.1, "Measuring and Reporting of Radioactivity in the Environs of Nuclear Power Plants". Changes in the reporting requirements which occur as a result of revisions to the above Regulatory Guides shall be incorporated into the reporting information no later than one reporting period following the reporting period in which the revision is issued. Submission of FSAR changes as stated in Regulatory Guide 1.16 shall not be required.

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

## INTRODUCTION

By a letter dated May 30, 1973, Vermont Yankee Nuclear Power Corporation (VYNPC) proposed changes to the Technical Specifications of Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would correct errors and inadequacies. As discussed in our Safety Evaluations for Change No. 13 dated January 17, 1974, and Change No. 15 dated January 28, 1974, to the Technical Specifications of Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station, the reissuance of the Technical Specifications will be made by sections. This Safety Evaluation will complete this review of the Technical Specifications and will review "Limiting Conditions for Operation" and "Surveillance Requirements" for Sections 3.8 and 4.8, "Radioactive Effluents", Sections 3.9 and 4.9, "Radioactive Effluent Monitoring Systems", Sections 3.10 and 4.10, "Auxiliary Electrical Power Systems", and Sections 3.11 and 4.11, "Refueling", Section 5.0, "Design Features", and "Administrative Controls" for Section 6.1, "Organization", Section 6.2, "Review and Audit", Section 6.3, "Action to be Taken in the Event of an Abnormal Occurrence in Plant Operation", Section 6.4, "Action to be Taken if a Safety Limit is Exceeded", Section 6.5, "Plant Operating Procedures", Section 6.6, "Plant Operating Records", and Section 6.7, "Plant Reporting Requirements".

## DISCUSSION

### Sections 3.8 and 4.8

Major modifications and organization have been made to Sections 3.8 and 4.8. The original Sections 3.8 and 4.8 have been separated into two different categories which address the radioactive effluents limits and their surveillance in this section and the radioactive effluent monitoring systems and their surveillance in Sections 3.9 and 4.9. The radioactive effluent limits have been evaluated in accordance with

Regulatory requirements. The liquid effluents remain limited on the basis of 10 CFR Part 20 requirements. However, action levels and reporting levels have been applied on the basis of meeting the "as low as practicable" concept which represents a few percent of the 10 CFR Part 20 limits. The surveillance of the liquid effluents remains consistent with the requirements for measurements needed to meet reporting requirements. The radioactive liquid storage limits have been reevaluated on the basis of meeting Regulatory requirements for radioactive waste treatment systems as stated in Regulatory Guide 1.29, "Seismic Design Classification". The surveillance requirements are consistent with controlling the quantity of radioactive liquid storage through sampling and recycling. The basis has been rewritten to establish the approach taken. The gaseous effluent limits for the gross radioactivity have been reevaluated on the basis of the site meteorological data and a finite cloud dose analysis. Action levels and reporting levels have been applied to meet the "as low as practicable" concept by using fractions of the 10 CFR Part 20 limits established by the Staff's evaluation. With the installation and operation of the augmented off-gas (AOG) system, meeting the "as low as practicable" levels should not restrict reactor operation. The gaseous effluent limits for radioiodine 131 have been evaluated on the basis of Appendices B and C of Regulatory Guide 1.42, "Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors". Allowance has not been given for the real cow that results in the highest milk-child thyroid dose since the cow population distribution data were not available. The gaseous effluent limits for radioactive particulates with half-lives greater than 8 days have been evaluated on the basis of the site meteorological data and 10 CFR Part 20 limits. Action levels and reporting levels consistent with the "as low as practicable" concept have been established for radioiodine and radioactive particulates. The surveillance requirements for all gaseous effluents have been established to be consistent with Regulatory Guide 1.21, "Measuring and Reporting of Effluents from Nuclear Power Plants". The Bases have been rewritten to describe the approach taken.

Sections 3.9 and 4.9

Separating the monitoring systems and their surveillance from the radioactive effluent limits and their surveillance has allowed the use of limiting conditions for operation with respect to the monitoring systems. The liquid effluent monitoring system is not changed nor is its surveillance. The radioactive liquid storage does not have a continuous monitoring system. The gaseous effluent monitoring system consists of the system in the plant stack which monitors the gross radioactivity, radioiodine, and the radioactive particulates and the systems located throughout the augmented off-gas system which monitor hydrogen and gross radioactivity. Only the latter system can cause an alarm and/or isolation of the releases. The surveillance requirements establish the frequency for checks, functional tests, and calibrations necessary to assure proper operation of the monitoring equipment. Vermont Yankee has agreed to perform a test program on the charcoal cartridges used to determine the radioiodine releases and the HEPA filters used to determine the radioactive particulate releases to evaluate the collection efficiency of these cartridges and filters under operational conditions and compare their results to those provided by the supplier. Tests will be performed on several cartridges and filters from each batch received at the facility for monitoring use. Results of this program will be provided to the Regulatory staff as a special report. The liquid process monitors consist of a gross radioactivity monitor to detect possible leaks of radioactivity into the service water and cooling tower water. The surveillance requirements are consistent with proper operation of the monitoring equipment. The offsite radiological monitoring program has been placed in this section with some minor corrections. This program with modifications will be transferred from the Appendix A Technical Specifications to the Appendix B Technical Specifications as soon as the Appendix B specifications are revised and reissued.

Sections 3.10 and 4.10

Only minor word changes and reorganization of these sections were made for clarification.

Sections 3.11 and 4.11

Major reorganization of these sections was made without changing the intent or conditions for refueling. Specifications 3.11.F and 4.11.F with Bases were added to indicate the conditions assumed for the safety analysis of a refueling accident.

Section 5.0

No changes were made in this section.

Section 6.0

Subsection 6.1 was updated to reflect the current Vermont Yankee organization. Subsection 6.2 was changed to show the combining of the position for Reactor and Computer Supervisor. No changes were made in Subsections 6.3 and 6.4. Subsection 6.5 was changed to include the respiratory protection section on respiratory protective equipment which had been Subsection 6.8. Specification 6.5.B.4 was added to Subsection 6.5 after inadvertently being left out in the issuance of Technical Specifications dated March 21, 1972, as discussed in the Vermont Yankee submittal dated May 30, 1973. Subsection 6.6 was unchanged except for the addition of three records required to be retained by Regulatory requirements. Subsection 6.7 was rewritten to establish reporting requirements which are consistent with Regulatory Guides pertaining to this subject. These Regulatory Guides have been listed in the specification.

## CONCLUSIONS

On the basis of our evaluation, we have concluded that the changes proposed by VYNPC, as modified, and the changes necessary to meet Regulatory requirements do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The Technical Specifications should be reissued as proposed by Vermont Yankee and modified by the Regulatory staff for Sections 3.8, 3.9, 3.10, 3.11, 4.8, 4.9, 4.10, 4.11, 5.0, 6.0, 6.1, 6.2, 6.3, 6.4, 6.5, 6.6, and 6.7 including Bases. These sections complete the reissuance of Appendix A Technical Specifications.

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