A-40 11.21.77

1. If, during power operation, an unexpected failure results in a complete loss of coolant tower system, the above closed cycle restriction may be modified to permit an orderly shutdown using the main condenser as a heat sink in the open cycle mode. In this event, the plant shall be reduced below 25 percent power operation as rapidly as possible and shutdown within twenty-four hours.

2. Vermont Yankee will define a comprehensive environmental (chemical, biological, and thermal) monitoring program for inclusion in the Technical Specifications, which is acceptable to the Commission for determining changes which many occur in land and water ecosystems as a result of plant operation.

3. If harmful effects or evidence of irreversible damage in land or water ecosystems, as a result of facility operation are detected by the monitoring program, Vermont Yankee shall provide an analysis of the problem to the Commission and to the advisory group for the Technical Specifications, and Vermont Yankee thereafter will provide, subject to the review by the aforesaid advisory group, a course of action to be taken immediately to alleviate the problem.

4. Vermont Yankee will grant authorized representatives of the Massachusetts Department of Public Health (MDPH) and Metropolitan District Commission (MDC) access to records and charts related to discharge of radioactive materials to the Connecticut River.

5. Prior to discharge of each tank (batch) of liquid radioactive effluents, a representative sample thereof shall be collected and held for independent analysis by the Commonwealth of Massachusetts. Authorized representatives of the Commonwealth shall pick up such samples at the plant site.

6. Vermont Yankee will furnish advance notification of each schedule calibration of liquid effluent monitors to MDPH and MDC and, upon request, will permit authorized representatives of the Commonwealth of Massachusetts to be present during such calibrations.

7. Vermont Yankee will permit authorized representatives of the MDPH and MDC to examine the chemical and radioactivity analyses performed by Vermont Yankee.

8. Vermont Yankee shall immediately notify MDPH, or an agency designated by MDPH, in the event concentrations of radioactive materials in liquid effluents, measured at the point of release from Vermont Yankee, exceed the limit set forth in the facility Technical Specifications, Appendix A, paragraph 3.8.A.1. Vermont Yankee will also notify MDPH in writing within 30 days following the release of radioactive materials in liquid effluents in excess of 10 percent of the limit set forth in the facility Technical Specifications, Appendix A, paragraph 3.8.A.1.

9. A report shall be submitted to MDPH and MDC within sixty days of January 1st and July 1st of each year of plant operation, specifying the total quantities of radioactive materials released to the Connecticut River during the previous six months. The report shall contain the following information:

   (a) Total curie activity discharged other than tritium and dissolved gases.

   (b) Total curie alpha activity discharged.

   (c) Total curies of tritium discharged.

   (d) Total curies of dissolved radio-gases discharged.

   (e) Total volume (in gallons) of liquid waste discharged.

(f) Total volume (in gallons) of dilution water.

(g) Average concentration at discharge outfall.

(h) Time, date, and duration of maximum concentration released (average over the period of release).

(i) Total radioactivity (in curies) released by nuclide including dissolved radio-gases.

(j) Percent of Technical Specification limit for total activity released.

10. Upon notification by MDPH or MDC that all plans and construction for the diversion of water from the Connecticut River to recharge Quabbin Reservoir have been completed, Vermont Yankee shall establish a system of communication and notification, satisfactory to MDPH and MDC, to give adequate warning to the appropriate agency or agencies of the Commonwealth of Massachusetts of any accidental discharge of radioactive materials into the Connecticut River from the facility.

11. Upon notification in writing by MDPH or MDC that water from the Connecticut River is being diverted to recharge Quabbin Reservoir, Vermont Yankee shall submit to both MDPH and MDC, until receipt of notification that such diversion has been terminated, monthly reports of liquid radioactive releases.

12. Vermont Yankee shall establish and maintain a system of emergency cnotification to the states of Vermont and New Hampshire, and the Commonwealth of Massachusetts, satisfactory to the appropriate public health and public safety officials of those states and the Commonwealth, which provides for:

(a) Notice of site emergencies as well as general emergencies.

(b) Direct microwave communication with the state police headquarters of the respective states and the Commonwealth when the transmission facilities of the respective states and the Commonwealth so permit, at the expense of Vermont Yankee.

(c) A verification or coding system for emergency messages between Vermont Yankee and the state police headquarters of the respective states and the Commonwealth.

13. Vermont Yankee shall furnish advance notification to MDPH, or to another Commonwealth agency designated by MDPH, of the time, method, and proposed route through the Commonwealth of any shipments of nuclear fuel and wastes to and from the Vermont Yankee facility which will utilize railways or roadways in the Commonwealth.

F. The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1 through 3.1.20 of the NRC's Fire Protection Safety Evaluation (SE) on the facility dated January 13, 1978. These modifications shall be completed as specified in Table 3.1 of the SE. In addition, the licensee shall submit the additional information identified in Table 3.2 of this SE in accordance with the schedule contained therein. In the event these dates for submittal cannot be met, the licensee shall submit a report, explaining the circumstances, together with a revised schedule.

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1.13.78

3.G. <u>Security Plan</u>

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10CFR73.55 (51FR27817 and 27822) and to the authority of 10CFR50.90 and 10CFR50.54(p). The plans, which contain Safeguards Information protected under 10CFR73.21, are entitled: "Vermont Yankee Nuclear Power Station Physical Security Plan," with revisions submitted through March 16, 1988; "Vermont Yankee Nuclear Power Station Training and Qualification Plan," with revisions submitted through November 10, 1982; and "Vermont Yankee Nuclear Power Station Safeguards Contingency Plan," with revisions submitted through December 30, 1985. Changes made in accordance with 10CFR73.55 shall be implemented in accordance with the schedule set forth therein.

A-107
8.25.88
10.20.88

3.H. This paragraph deleted.

4. This license is effective as of the date of issuance and shall expire at midnight on December 11, 2007.

FOR THE ATOMIC ENERGY COMMISSION

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosures:
Appendices A & B - Technical Specifications

Date of Issuance: [illegible]

TABLE 3.2.1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Core Spray - A & B (Note 1)

| Minimum Number of Operable Instrument Channels per Trip System | Trip Function | Trip Level Setting | Required Action When Minimum Conditions for Operation are Not Satisfied |
|---|---|---|---|
| 2 | High Drywell Pressure | $\leq$2.5 psig | Note 2 |
| 2 | Low-Low Reactor Vessel Water Level | $\geq$82.5" above top of enriched fuel | Note 2 |
| 1 | Low Reactor Pressure (PT-2-3-56C/D(S1) | 300 $\leq$ P $\leq$ 350 psig | Note 2 |
| 2 | Low Reactor Pressure (PT-2-3-56A/B(S1) & 52C/D(M)) | 300 $\leq$ P $\leq$ 350 psig | Note 2 |
| 1 | Time Delay (14A-K16A & B) | $\leq$10 seconds | Note 2 |
| 2 | Pump 14-1A, Discharge Pressure | $\geq$100 psig | Note 5 |
| 1 | Auxiliary Power Monitor | — | Note 5 |
| 1 | Pump Bus Power Monitor | — | Note 5 |
| 1 | High Sparger Pressure | $\leq$5 psid | Note 5 |
| 1 | Trip System Logic | — | Note 5 |

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

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

TABLE 3.2.5 NOTES

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

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

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

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

5. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to $48 \times 10^6$ lbs/hr core flow. Refer to the Core Operating Limits Report for acceptable values for N. $\Delta W$ is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

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

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

8. With the number of operable channels less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

9. With one RBM channel inoperable:

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

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

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

TABLE 3.2.6

POST-ACCIDENT INSTRUMENTATION
(continued)

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TABLE 3.2.6 NOTES

Note 1 - From and after the date that a parameter is reduced to one indication, operation is permissible for 30 days. If a parameter is not indicated in the Control Room, continued operation is permissible during the next seven days. If indication cannot be restored within the next six hours, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following 18 hours.

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

Note 3 - From and after the date that this parameter is reduced to one indication in the Control Room, continued reactor operation is permissible during the next 30 days. If both channels are inoperable and indication cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following 18 hours.

Note 4 - From and after the date that safety/relief valve position from pressure switches is unavailable, reactor operation may continue provided safety/relief valve position can be determined from Recorder #2-166 (steam temperature in SRVs, 0-600°F) and Meter 16-19-33A or C (torus water temperature, 0-250°F). If both parameters are not available, the reactor shall be in a hot shutdown condition in six hours and a cold shutdown condition in the following 18 hours.

Note 5 - From and after the date that safety valve position from the acoustic monitor is unavailable, reactor operation may continue provided safety valve position can be determined from Recorder #2-166 (thermocouple, 0-600°F) and Meter #16-19-12A or B (containment pressure 0-275 psia). If both indications are not available, the reactor shall be in a hot shutdown condition in six hours and in a cold shutdown condition in the following 18 hours.

Note 6 - Within 30 days following the loss of one indication, or seven days following the loss of both indications, restore the inoperable channel(s) to an operable status or a special report to the Commission pursuant to Specification 6.7 must be prepared and submitted within the subsequent 14 days, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status.

Note 7 - From and after the date that this parameter is unavailable by Control Room indication, and cannot be restored within 24 hours, continued reactor operation is permissible for the next 30 days provided that local sampling capacity is available. If the Control Room indication cannot be restored within 30 days, the reactor shall be in hot shutdown within six hours and in cold shutdown within the subsequent 24 hours.

3.6 LIMITING CONDITIONS FOR OPERATION

## 3.6 REACTOR COOLANT SYSTEM

### B. Coolant Chemistry

1. a. During reactor power operation, the radioiodine concentration in the reactor coolant shall not exceed 1.1 microcuries of I-131 dose equivalent per gram of water, except as allowed in Specification 3.6.B.1.b.

4.6 SURVEILLANCE REQUIREMENTS

## 4.6 REACTOR COOLANT SYSTEM

Specification

### A. Pressure and Temperature Limitations (cont.)

5. The reactor vessel irradiation surveillance specimens shall be removed and examined to determine changes in material properties in accordance with the following schedule:

| CAPSULE | REMOVAL YEAR |
|---|---|
| 1 | 10 |
| 2 | 30 |
| 3 | Standby |

The results shall be used to update Figures 3.6.2 and 3.6.3. The removal times shall be referenced to the refueling outage following the year specified, referenced to the date of commercial operation.

### B. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when steam jet air ejector monitors indicate an increase in radioactive gaseous effluents of 25 percent or 5000 uCi/sec, whichever is greater,

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TABLE 4.7.2.a

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

| Isolation Group (1) | Valve Identification | Number of Power Operated Valves Inboard | Number of Power Operated Valves Outboard | Maximum Operating Time (sec) | Normal Position | Action on Initiating Signal |
|---|---|---|---|---|---|---|
| 1 | Main Steam Line Isolation (2-80A, D & 2-86A, D) | 4 | 4 | 5(Note 2) | Open | GC |
| 1 | Main Steam Line Drain (2-74, 2-77) | 1 | 1 | 35 | Closed | SC |
| 1 | Recirculation Loop Sample Line (2-39, 2-40) | 1 | 1 | 5 | Closed | SC |
| 2 | RHR Discharge To Radwaste (10-57, 10-66) | | 2 | 25 | Closed | SC |
| 2 | Drywell Floor Drain (20-82, 20-83) | | 2 | 20 | Open | GC |
| 2 | Drywell Equipment Drain (20-94, 20-95) | | 2 | 20 | Open | GC |
| 3 | Drywell Air Purge Inlet (16-19-9) | | 1 | 10 | Closed | SC |
| 3 | Drywell Air Purge Inlet (16-19-8) | | 1 | 10 | Open | GC |
| 3 | Drywell Purge & Vent Outlet (16-19-7A) | | 1 | 10 | Closed* | SC |
| 3 | Drywell Purge & Vent Outlet Bypass (16-19-6A) | | 1 | 10 | Closed | SC |
| 3 | Drywell & Suppression Chamber Main Exhaust (16-19-7) | | 1 | 10 | Closed* | SC |
| 3 | Suppression Chamber Purge Supply (16-19-10) | | 1 | 10 | Closed | SC |
| 3 | Suppression Chamber Purge & Vent Outlet (16-19-7B) | | 1 | 10 | Closed | SC |
| 3 | Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B) | | 1 | 10 | Open | GC |
| 3 | Exhaust to Standby Gas Treatment System (16-19-6) | | 1 | 10 | Open | GC |
| 3 | Containment Purge Supply (16-19-23) | | 1 | 10 | Open | GC |
| 3 | Containment Purge Makeup (16-20-20, 16-20-22A, 16-20-22B) | | 3 | NA | Closed | SC |
| 5 | Reactor Cleanup System (12-15, 12-18) | 1 | 1 | 25 | Open | GC |
| 5 | Reactor Cleanup System (12-68) | | 1 | 45 | Open | GC |
| 6 | HPCI (23-15, 23-16) | 1 | 1 | 55 | Open | GC |
| 6 | RCIC (13-15, 13-16) | 1 | 1 | 20 | Open | GC |
| | Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B) | | 2 | NA | Closed | SC |
| | Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B) | | 2 | NA | Closed | Process |
| | Control Rod Hydraulic Return Check Valve (3-181) | | | NA | Open | Process |
| 3 | Containment Air Sampling (VG 23, VG 26, 109-76A&B) | | 4 | 5 | Open | GC |

*Valves 16-19-7 and 16-19-7A shall have stops installed to limit valve opening to 50° or less.

4.7.A (Continued)

The design pressure of the drywell and absorption chamber is 56 psig.(2) The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.5%/day at 44 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90% for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 1.65 rem and the maximum total thyroid dose is about 280 rem at the site boundary over an exposure duration of two hours. The resultant dose that would occur for the duration of the accident at the low population distance of 5 miles is lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (L a) to a value of 0.80%/day.

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

To allow a margin for possible leakage deterioration between test intervals, the maximum allowable operational leak rate (Ltm), which will be met to remain on the normal test schedule, is 0.75 Lt.

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

(2) 62 psig is the maximum allowable peak accident pressure for this design (56 psig) pressure.

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

5. The Radiation Protection Supervisor or Plant Health Physicist shall meet or exceed the qualifications of Regulatory Guide 1.8, Revision 1 (September 1975).

6. The Shift Engineer shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

7. If the Operations Supervisor does not possess a Senior Operator License, then an Assistant Operations Supervisor shall be designated that does possess a Senior Operator License. All instructions to the shift crews involving licensed activities shall then be approved by designated Assistant Operations Supervisor.

8. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate on-site manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

E. A Fire Brigade of at least 5 members shall be maintained on-site at all times.# This excludes 2 members of the minimum shift crew necessary for safe shutdown of the plant and any personnel required for other essential functions during a fire emergency.

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# Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.