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Docket No. 50-271

FEBRUARY 2 8 1979

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
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Dear Mr. Groce:

By letter dated January 31, 1979 we transmitted Amendment No. 50 to Facility Operating License No. DPR-28, for the Vermont Yankee Nuclear Power Station, relating to the Mark I Containment Short Term Program. Certain Technical Specification pages necessary for implementation of Amendment No. 50 were inadvertently omitted.

Please substitute the enclosed instruction sheet and Technical Specification pages for those transmitted with Amendment No. 50.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Technical Specification
pages

cc w/enclosure:
See page 2

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ATTACHMENT TO LICENSE AMENDMENT NO. 50

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix a Technical Specifications as follows:

Remove	Insert
34	34
49	49
60	60
126	126
126a	126a
129	129
-	129a
133	133
135	135
138	138
139	139
142	142

Changes on the revised pages are shown by marginal lines.

F. Mechanical Vacuum Pump Isolation

1. Whenever the main steam line isolation valves are open, the mechanical vacuum pump shall be capable of being automatically isolated and secured by a signal of high radiation in the main steam line tunnel or shall be manually isolated and secured.

2. If Specification 3.2.F.1 is not met following a routine surveillance check, the reactor shall be in the cold shutdown within 24 hours.

G. Post-Accident Instrumentation

During reactor power operation, the instrumentation that displays information in the control room necessary for the operator to initiate and control the systems used during and following a postulated accident of abnormal operating condition shall be operable in accordance with Table 3.2.6.

H. Drywell to Torus ΔP Instrumentation

1. During reactor power operation, the Drywell to Torus ΔP Instrumentation (recorder #1-156-3 and instrument DPI-1-158-6) shall be operable except as specified in 3.2.H.2.

2. From and after the date that one of the drywell to torus ΔP instruments is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless the instrument is sooner made operable. If both instruments are made or found to be inoperable, and indication cannot be restored within a six hour period, 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 eighteen hours.

F. Mechanical Vacuum Pump Isolation

During each operating cycle, automatic isolation and securing of the mechanical vacuum pump shall be verified while the reactor is shutdown.

G. Post-Accident Instrumentation

The post-accident instrumentation shall be functionally tested and calibrated in accordance with Table 4.2.6.

H. Drywell to Torus ΔP Instrumentation

The Drywell to Torus ΔP Instrumentation shall be calibrated once every six months and an instrument check will be made once per shift.

TABLE 3.2.6

POST-ACCIDENT INSTRUMENTATION

Minimum Number of Operable Instrument Channels	Parameter	Type of Indication	Instrument Range
2	Drywell Atmospheric Temperature (Note 1)	Recorder #16-19-45 Recorder #TR-1-149	0-300°F 0-300°F
2	Drywell Pressure (Note 1) Torus Pressure (Note 1)	Recorder #16-19-44	0-80 psia 0-80 psia
2	Torus Water Level (Note 3)	Meter #16-19-46A Meter #16-19-46B	0-3 ft. 0-3 ft.
2	Torus Water Temperature (Note 1)	Meter #16-19-48	60-180°F
2	Reactor Pressure (Note 1)	Recorder #6-97 Meter #6-90A Meter #6-90B	0-1200 psig 0-1200 psig 0-1200 psig
2	Reactor Vessel Water Level (Note 1)	Meter #2-3-91A Meter #2-3-91B	(-150)-0-(+150)"H_2O (-150)-0-(+150)"H_2O
1	Control Rod Position (Note 1,2)	Meter	0-48" RPIS
1	Neutron Monitor (Note 1,2)	Meter	0-125% Rated Flux
1	Torus Air Temperature (Note 1)	Recorder #TR-16-19-45	0-300°F

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

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

Note 3 - From and after the date that this parameter is reduced to one indication in the control room, continuted reactor operation is permissible during the next thirty 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.

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TABLE 4.2.6

CALIBRATION FREQUENCIES

POST-ACCIDENT INSTRUMENTATION

Parameter	Calibration	Instrument Check
Drywell Atmosphere Temperature	every 6 months	once each day
Drywell and Torus Pressure	every 6 months	once each day
Torus Water Level	every 6 months	once each shift
Torus Water Temperature	every 6 months	once each day
Reactor Pressure	every 6 months	once each day
Reactor Vessel Water Level	every 6 months	once each day
Control Rod Position	(note 5)	once each day
Neutron Monitor	Same as reactor protection systems	once each day
Torus Air Temperature	every 6 months	once each day

3.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

   a. Maximum Water Temperature during normal operation - 90°F.

   b. Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F and shall not be above 90°F for more than 24 hours.

   c. If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 90°F.

   d. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temperature excees 120°F.

4.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per shift. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 LIMITING CONDITIONS FOR OPERATION

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

4.7 SURVEILLANCE REQUIREMENTS

4.7 STATION CONTAINMENT SYSTEMS

2. The primary containment integrity shall be demonstrated as required by Appendix J to 10 CFR Part 50. The primary containment shall meet the containment acceptance requirements set forth in that appendix.

a. Penetrations and seals listed in Table 4.7.1 shall be leak tested at 44 psig (Pa).

b. Type C tests shall be performed on the isolation valves listed in Table 4.7.2.a.

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

7. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4 percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.7.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

8. If Specification 3.7.A.1 through 3.7.A.7 cannot be met, an orderly shutdown shall be initiated immediately and the reactor shall be in a cold shutdown condition within 24 hours.

(4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice.

7. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITIONS FOR OPERATION

9. Drywell/Suppression Chamber d/p

a. Differential pressure between the drywell and suppression chamber shall be maintained $\geq$1.7 psi except as specified in 3.7.A.9.b and 3.7.A.9.c below.

b. The $\geq$ 1.7 psi differential pressure shall be established within 24 hours of achieving operating pressure and temperature. The differential pressure may be reduced to < 1.7 psi 24 hours prior to commencing a cold shutdown.

c. The differential pressure may be reduced to < 1.7 psi for a maximum of four hours (period to begin when the ΔP is reduced to < 1.7) during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-suppression chamber vacuum breakers, and the suppression chamber-reactor building vacuum breakers, and SBGTS testing.

d. If the specifications of 3.7.A.9.a cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

4.7 SURVEILLANCE REQUIREMENTS

9. Drywell/Suppression Chamber d/p

a. The differential pressure between the drywell and suppression chamber shall be recorded once per shift.

b. The operability of the low differential pressure alarm shall be verified once per week.

3.7 LIMITING CONDITIONS FOR OPERATION

4.7 SURVEILLANCE REQUIREMENTS

(2) The instrument line flow check valves shall be tested for proper operation.

b. At least once per quarter:

(1) All normally open power-operated isolation valves (except for main steam isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 50 percent of rated, trip main steam isolation valves (one at a time) and verify closure time.

c. At least twice per week:

(1) The main steamline isolation valves shall be exercised by partial closure and subsequent reopening.

2. In the event any isolation valve specified in Table 4.7.2 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Whenever an isolation valve listed in 4.7.2 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

TABLE 4.7.2.a.

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 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

Bases:

3.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment

The integrity of the primary containment and operation of the core standby cooling systems in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical, above atmospheric pressure and temperature above 212°F. An exception is made to this requirement during initial core loading and while a low power test program is being conducted and ready access to the reactor vessel is required. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to less than 1.30% delta k.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the allowable pressure suppression chamber pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber (Reference Section 5.2 FSAR).

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 44 psig, which is below the design of 56 psig.[3] The minimum volume of 68,000 ft^3 results in a submergency of approximately four feet. The majority of the Bodega tests[2] were run with a submerged length of four feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

The maximum temperature at the end of blowdown tested during the Humboldt Bay[1] and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperature above 170°F.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (see Vermont Yankee letter dated September 13, 1976) which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure of 1.7 psid and a suppression chamber water level corresponding to a downcomer submergence range of 4.29 to 4.54 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3.7.A (cont'd)

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 210°F prior to the DBA-LOCA. Maintaining a pool temperature of 90°F will assure that the 170°F limit is not approached.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Double isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the structural integrity of the containment is maintained.

Technical Specification 3.7.A.9.c is based on the assumption that the operability testing of the pressure suppression chamber-reactor building vacuum breaker, when required, will normally be performed during the same four hour testing interval as the pressure suppression chamber-drywell vacuum breakers in order to minimize operation with <1.7 psi, differential pressure.

The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure is 2 psig.

The capacity of the ten (10) drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows eight (8) operable

4.7.A (cont'd)

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 over a 30-day period. Thus, these doses are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines. An additional factor of two for conservatism is added to the above doses by limiting the test leak rate (La) to a value of 0.80%/day.

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

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

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.

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