1/31/79

Docket No. 50-271

Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

In response to your requests for license amendment dated December 10, 1976, April 14, 1977 and May 16, 1978, the Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station.

This amendment incorporates provisions into the facility Technical Specifications which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that facility operation will be in accordance with the assumptions utilized in your facility's plant-unique analysis which was performed in conjunction with the Mark I Containment Short Term Program evaluation.

The suppression chamber differential pressure control method which this amendment applies to, makes use of continuous purge from the torus to the Reactor Building vent. Although we discourage the use of continuous containment purging, we have concluded that this particular application is provisionally acceptable based on the considerations described in the enclosed Safety Evaluation and because this feature is desirable from an operational standpoint. In a separate letter dated November 29, 1978, we requested information relating to the overall practice of containment purging at your facility. Our review of your response to this letter and our review of your compliance with the requirements of 10 CFR Part 50, Appendix I may require a modification to the differential pressure control purging subsystem.

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OFFICE ➤						
SURNAME ➤						
DATE ➤						

The enclosed license amendment reflects those changes to your original request for license amendment which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated Februray 13, 1976, relative to Facility Operating License No. DPR-28 is terminated.

Copies of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 50 to DPR-28
2. Safety Evaluation
3. Notice

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Copies of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

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

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DATE	7/28/78	7/26/78	7/31/78	7/ /78	7/ /78	7/ /78

cc: Mr. S. D. Karpyak
Vermont Yankee Nuclear Power
Corporation
77 Grove Street
Rutland, Vermont 05701

Mr. Donlad E. Vandenburgh
Vice President
Vermont Yankee Nuclear Power
Corporation
Turnpike Road, Route 9
Westboro, Massachusetts 01581

John A. Ritsher, Esquire
Rope & Gray
225 Franklin Street
Boston, Massachusetts 02110

Laurie Burt
Assistant Attorney, General
Environmental Protection Division
Attorney General's Office
One Ashburton Place, 19th Floor
Boston, Massachusetts 02108

New England Coalition on Nuclear
Pollution
Hill and Dale Farm
West Hill - Faraway Road
Putney, Vermont 05346

Mr. Raymond H. Puffer
Chairman
Board of Selectman
Vermon, Vermont 05354

W. F. Conway, Plant Superintendent
Vermont Yankee Nuclear Power
Corporation
P. O. Box 157
Vernon, Vermont 05354

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

John R. Stanton, Director
Radiation Control Agency
Hazen Drive
Concord, New Hampshire 03301

John W. Stevens
Conservation Socity of Southern
Vermont
P. O. Box 256
Townshend, Vermont 05353

Mr. Davis M. Scott
Radiation Health Engineer
Agency of Human Services
Division of Occupational Health
P. O. Box 607
Barre, Vermont 05641

Richard E. Ayres, Esquire
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Honorable M. Jerome Diamond
Attorney General
John A. Calhoun
Assistant Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Yankee Atomic Electric Company - -

cc: Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection
Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region I Office
ATTN: EIS COORDINATOR
JFK Federal Building
Boston, Massachusetts 02203

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. DPR-28

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

   A. The applications for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated December 10, 1976, April 14, 1977, and May 16, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 31, 1979

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
129	129
-	129a
138	138
139	139

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 the 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 DP1-1-158-6) 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 during the next seven days. If reduced to one indication of a parameter operation is permissible for 30 days.

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

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 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.

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

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.

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.7.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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Introduction

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Vermont Yankee Nuclear Power Corporation submitted a Plant Unique Analysis (PUA) for the Vermont Yankee Nuclear Station. This analysis was performed to confirm the structural and functional capability of the containment suppression chamber and attached piping to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the NRC staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report", NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

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As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrodynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating this facility with differential pressure control to enhance the safety margins of the containment structure since early 1976 as a result of an Order issued by the NRC staff on February 13, 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water level, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting conditions for operation and surveillance requirements. Because these requirements replace the requirements of the February 13, 1976 Order, the Order is vacated.

By letters dated December 10, 1976, April 14, 1977 and May 16, 1978, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows.

Evaluation

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level will be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided, such that a comparison of the readings will indicate when one of the channels is inoperative or drifting. The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid and a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with differential pressure and torus water level.

There are certain periods during normal plant operations when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure/control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

1. <u>Startup and Shutdown</u>

   During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 24 hours following startup and 24 hours prior to a shutdown. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only a part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients, and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event to which differential pressure control is important to assure containment integrity (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

2. <u>Testing and Maintenance</u>

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure to permit the vacuum breakers to open. Because of the configuration at Vermont Yankee, testing of the suppression chamber-reactor building vacuum breakers and the standby gas treatment system may result in a reduction of drywell-torus differential pressure. For testing of high-energy systems (e.g., high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e., pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct these tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Specifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecesssary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control, i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown. The allowable time to conduct a controlled shutdown has been minimized, because the containment transient response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment integrity, and the periods of time differential pressure control is completely removed are acceptably small.

The torus pumpback system will exhaust about 1 cfm continuously from the torus to the Reactor Building vent to maintain the required torus-drywell pressure difference. The licensee predicted the potential exclusion area boundary dose resulting from pumpback venting prior to containment isolation after a design basis LOCA would increase only a fraction of a rem (the resulting dose would remain well within the 10 CFR 100 guidelines).

Releases from accidents inside containment will be limited by containment isolation, which also isolates the torus pumpback vent lines. In addition, the torus air removed by the pumpback system is released through the Reactor Building vent, which is monitored by radiation detectors and closed on a high radiation signal from those detectors. We have performed an independent analysis of the effect of the proposed torus pumpback system on the potential dose consequences of a LOCA. Our assumptions for that analysis and the resulting potential dose consequences are presented in Tables 1 and 2. We presented potential dose consequences of 148 Rem to the thyroid and 5.7 Rem to the whole body for the LOCA in the Safety Evaluation for Vermont Yankee dated June 1971.

The potential dose consequences presented in Table 2 represent an insignificant increase in the consequences presented above, and the resulting potential dose consequences are still well within the guidelines of 10 CFR Part 100.

Routine flow from the torus will release some radioactivity to the environment because torus air contains some radioactivity after certain reactor operations, such as a safety-relief valve lift. The licensee has estimated that during normal operation, including anticipated operational occurrences, 625 microcuries of I-131 per year will be released from the torus pumpback system. The licensee based this estimate on measurements of the torus airborne radioactivity. Based on cost-benefit analyses, the licensee found that the cost of installing and maintaining a charcoal filter in the torus vent line would exceed the benefit to the population from reduced radioiodine exposure.

We have reviewed the bases and methods the licensee used to estimate the routine radioactivity releases from the torus pumpback system. Based on that review and on our knowledge of effluents from boiling water reactors, we conclude that the licensee's estimate of 625 microcuries of I-131 per year is reasonable. We have also reviewed the licensee's cost-benefit analysis for adding a charcoal filter to the torus vent line. We have performed an independent analysis using more conservative assumptions, and we agree with the licensee's conclusion that this would not be cost-beneficial to the public. In addition, the release limits given in the Technical Specifications for the Vermont Yankee station are not being changed by this action; hence, the releases from torus-pumpback system operation, when added to the other normal plant releases, must remain within current effluent limits.

We are currently evaluating Vermont Yankee, as well as all other operating light water reactors, to determine compliance of Vermont Yankee with the requirements of 10 CFR Part 50, Appendix I. At the conclusion of that evaluation, the Technical Specifications will be amended as necessary to implement Appendix I for the operation of Vermont Yankee. The releases of radioactivity from torus pumpback operation will be included in the evaluation and the implementing changes to the Technical Specifications.

On November 29, 1978, we requested information from the licensee related to purging during operation. The information that the licensee provides in response to this request will include consideration of the torus pumpback vent lines. We will review this information generically according to the guidelines described in our letter of November 29, 1978.

We believe the licensee has adequately considered the potential impact on the public of accidental and normal releases from the torus pumpback system venting and believe torus pumpback system operation will not pose an undue risk to the public health and safety.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusions

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 31, 1979

TABLE 1

ASSUMPTIONS FOR AND POTENTIAL CONSEQUENCES AT THE

EXCLUSION AREA BOUNDARY OF LOCA CONTRIBUTION BY TORUS

PUMPBACK SYSTEM FOR VERMONT YANKEE

Assumptions	Value
Valve Isolation Time	10.5 seconds
Primary Coolant Iodine Activity, Dose Equivalent I-131	1.1 microcuries per gram
0 - 2 hours X/Q Value, Exclusion Area Boundary (94 meter stack release)	4.0×10^{-4} seconds per cubic meter
Primary Coolant Released from Torus Pumpback System Prior to Isolation	350 pounds

TABLE 2

EXCLUSION AREA BOUNDARY (EAB) CONSEQUENCES

FROM ACCIDENT (CONTRIBUTION FROM TORUS PUMPBACK SYSTEM)

Doses, Rem	
Thyroid	Whole Body
< 1.0	< 0.1

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 50 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation (the licensee), which revised the Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility), located near Vernon, Vermont. The amendment is effective as of the date of its issuance.

The amendment revised the Technical Specifications to incorporate requirements for establishing and maintaining the drywell to suppression chamber differential pressure and suppression chamber water level, to maintain the margins of safety established in the NRC staff's "Mark I Containment Short Term Program Safety Evaluation", NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 FR 13118, March 29, 1978.

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

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The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) applications for amendment dated December 10, 1976, April 14, 1977 and May 16, 1978, (2) Amendment No. 50 to License No. DPR-28, (3) the Commission's related Safety Evaluation and (4) the Commission's Order for Modification of License dated February 13, 1976. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont. A single copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 31 day of January 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

TAIppolito

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