June 20, 1995

Mr. Donald A. Reid
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
Ferry Road
Brattleboro, VT 05301

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M90880)

Dear Mr. Reid:

The Commission has issued the enclosed Amendment No. 145 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated October 28, 1994.

The amendment removes the Neutron Monitoring System (NMS) and Control Rod Position instrumentation from the Vermont Yankee Technical Specifications for post-accident monitoring and incorporates unrelated administrative changes.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

Original signed by E. Trottier
for

Daniel H. Dorman, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 145 to DPR-28
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Daniel H. Dorman, Project Manager
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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated October 28, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

   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.

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:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 145 , 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Reg J Vinny for

Phillip F. McKee, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 20, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 145

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Appendix A, Technical Specifications, with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
53	53
55	55
70	70
74	74
78	78

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 #TR-16-19-45 (TE-16-19-30A) Meter #TI-16-19-30B	0-350°F 0-350°F
2	Containment Pressure (Note 1)	Meter #PI-16-19-12A Meter #PI-16-19-12B	(-15) -(+260) psig (-15) -(+260) psig
2	Torus Pressure (Note 1)	Meter #PI-16-19-36A Meter #PI-16-19-36B	(-15) -(+65) psig (-15) -(+65) psig
2	Torus Water Level (Note 3)	Meter #LI-16-19-12A Meter #LI-16-19-12B	0-25 ft. 0-25 ft.
2	Torus Water Temperature (Note 1)	Meter #16-19-33A Meter #16-19-33C	0-250°F 0-250°F
2	Reactor Pressure (Note 1)	Meter #PI-2-3-56A Meter #PI-2-3-56B	0-1500 psig 0-1500 psig
2	Reactor Vessel Water Level (Note 1)	Meter #2-3-91A Meter #2-3-91B	(-200)-0-(+200)" H_2O (-200)-0-(+200)" H_2O
2	Torus Air Temperature (Note 1)	Recorder #TR-16-19-45 (TE-16-19-34) Meter #TI-16-19-41	0-350°F 50-300°F
2/valve	Safety/Relief Valve Position From Pressure Switches (Note 4)	Lights (SRV 2-71-1, 2, 3 (A thru D))	Closed - Open

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 - Deleted.

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 (-15) -(+260) psig). 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.

TABLE 4.2.6

CALIBRATION REQUIREMENTS

POST-ACCIDENT INSTRUMENTATION

Parameter	Calibration	Instrument Check
Drywell Atmosphere Temperature	Every 6 Months	Once Each Day
Containment Pressure	Once/Operating Cycle	Once Each Day
Torus Pressure	Once/Operating Cycle	Once Each Day
Torus Water Level	Once/Operating Cycle	Once Each Day
Torus Water Temperature	Every 6 Months	Once Each Day
Reactor Pressure	Once/Operating Cycle	Once Each Day
Reactor Vessel Water Level	Once/Operating Cycle	Once Each Day
Torus Air Temperature	Every 6 Months	Once Each Day
Safety/Relief Valve Position	Every Refueling Outage (Note 9) (a Functional Test to be performed quarterly)	Once Each Day
Safety Valve Position	Every Refueling Outage (Note 9) (a Functional Test to be performed quarterly)	Once Each Day

TABLE 4.2 NOTES

1. Initially once per month; thereafter, a longer interval as determined by test results on this type of instrumentation.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart.

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.

4. This instrumentation is expected from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

5. Deleted.

6. Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibration shall be performed prior to or during each startup or controlled shutdown with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when instruments are required to be operable.

7. This instrumentation is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every three months.

8. Functional tests and calibrations are not required when systems are not required to be operable.

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

BASES: 3.2 (Cont'd)

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus control rod motion is prevented.

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since, for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. For a break or other event occurring outside the drywell, the Automatic Depressurization System is initiated on low-low reactor water level only after a time delay. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the Specification are adequate to ensure the above criteria are met. The Specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

The ADS is provided with inhibit switches to manually prevent automatic initiation during events where actuation would be undesirable, such as certain ATWS events. The system is also provided with an Appendix R inhibit switch to prevent inadvertent actuation of ADS during a fire which requires evacuation of the Control Room.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. Any one upscale trip or two downscale trips of either set of monitors will cause the desired action. Trip settings for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leave the Reactor Building via the normal ventilation stack but that all activity is processed by the standby gas treatment system. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation at a radiation level equivalent to the maximum site boundary dose rate of 500 mrem/year as given in Specification 3.8.E.1.a. The monitoring system in the plant stack represents a backup to this system to limit gross radioactivity releases to the environs.

The purpose of isolating the mechanical vacuum pump line is to limit release of radioactivity from the main condenser. During an accident, fission products would be transported from the reactor through the main steam line to the main condenser. The fission product radioactivity would be sensed by the main steam line radiation monitors which initiate isolation.

Post-accident instrumentation parameters for Containment Pressure, Torus Water Level, Containment Hydrogen/Oxygen Monitor, and Containment High-Range Radiation Monitor, are redundant, environmentally and seismically qualified instruments provided to enhance the operators' ability to follow the course of an event. The purpose of each of these instruments is to provide detection and measurement capability during and following an accident as required by NUREG-0737 by ensuring continuous on-scale indication of the following: containment pressure in the (-15) -(+260) psig range; torus water level in the 0 to 25 foot range (i.e., the bottom to 5 feet above the normal water level of the torus pool); containment hydrogen/oxygen concentrations (0 to 30% hydrogen and 0 to 25% oxygen); and containment radiation in the 1 R/hr to 10^7 R/hr gamma.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 145 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

Section 6.2 of Generic Letter (GL) 82-33 requested that licensees provide a report on their implementation of Regulatory Guide (R.G.) 1.97, Revision 2, and methods for complying with the Commission's regulations including supporting technical justification for any proposed deviations or alternatives. A large number of deviation requests were received regarding neutron flux monitoring instrumentation at boiling water reactor (BWR) facilities. These requests were initially denied.

In support of these requests, the BWR Owners Group (BWROG) submitted NEDO-31558, "Position on NRC Regulatory Guide 1.97, Revision 3, Requirements for Post-Accident Neutron Monitoring System." NEDO-31558 proposed alternate criteria for neutron flux monitoring instrumentation in lieu of the Category 1 criteria stated in R.G. 1.97.

The staff's evaluation of NEDO-31558 is contained in a safety evaluation (SE) dated January 13, 1993, which was forwarded to the Vermont Yankee Nuclear Power Corporation (the licensee) in a letter dated April 29, 1993. The staff concluded the Category 1 neutron flux monitoring instrumentation is not needed for existing BWRs to cope with a Loss-of-Coolant Accident, Anticipated Transient Without Scram, or other accidents that do not result in severe core damage conditions. In the letter of April 29, 1993, the staff requested that the licensee review its neutron flux monitoring instrumentation against the criteria of NEDO-31558 to determine whether it met the stated criteria. The staff also noted that since the neutron flux monitoring instrumentation is no longer considered to be Category 1 instrumentation as defined in R.G. 1.97, the licensee may request removal of that instrumentation from the post-accident monitoring technical specification (TS) for the Vermont Yankee Nuclear Power Station (VYNPS).

The licensee reported the results of its review in a letter dated December 1, 1993. The staff reviewed the licensee's submittal and concluded in a letter dated February 7, 1994, that the post-accident monitoring system at the VYNPS meets the criteria of NEDO-31558 and is, therefore, an acceptable alternative to the guidance in R.G. 1.97.

By letter dated October 28, 1994, the licensee submitted a request for changes to the Vermont Yankee Nuclear Power Station Technical Specifications (TS). The requested changes would remove the neutron monitoring system (NMS) and control rod position instrumentation from the Vermont Yankee Technical Specifications for post-accident monitoring and incorporates unrelated administrative changes.

2.0 EVALUATION

The licensee proposed to change Tables 3.2.6 and 4.2.6 and associated Notes to remove the NMS instrumentation from the post-accident monitoring TS. As discussed in Section 1.0 above, the staff has previously determined that the NMS instrumentation at VYNPS is not Category 1 instrumentation as defined in R.G. 1.97 and may be removed from the plant TS. Therefore this change is acceptable.

The licensee proposed to change Tables 3.2.6 and 4.2.6 and associated Notes to remove the control rod position instrumentation from the post-accident monitoring TS. This instrumentation is classified as Category 3 in R.G. 1.97 and was provided in the VYNPS TS to provide redundancy for the NMS instrumentation which would be removed from the TS by the proposed amendment. This change is therefore acceptable.

The licensee proposed to change Table 3.2.6 and associated Notes and Bases to revise the ranges for the Containment Pressure instruments from "0-275 psia" to "(-15) - (+260) psig" and for the Torus Pressure instruments from "0-80 psia" to "(-15) - (+65) psig." These changes do not alter the actual instrument ranges, but by revising the units of pressure amend the TS to be consistent with the plant's Final Safety Analysis Report, the licensee's R.G. 1.97 submittal and the actual indicated ranges of the installed instruments. This change is therefore acceptable, and the staff has no objection to the proposed change to the associated Bases.

The licensee proposed to change the identification numbers for "Safety/Relief Valve Position From Pressure Switches" in Table 3.2.6 to correct a typographical error. This change is administrative and is therefore acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (60 FR 24922). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Dorman

Date: June 20, 1995