August 4, 1988

Docket No. 50-271

Mr. R. W. Capstick
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Capstick:

SUBJECT: ISSUANCE OF AMENDMENT NO. 105 TO FACILITY OPERATING LICENSE NO. DRP-28 (TAC#66952) VERMONT YANKEE NUCLEAR POWER STATION

The Commission has issued the enclosed Amendment No. 105 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated December 9, 1987.

This amendment changes the Technical Specifications to accommodate limiting conditions of operation and surveillance requirements for the Automatic Depressurization System (ADS) because of modifications made in response to NUREG 0737, Item II. K.3.18.

A copy of our Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

Vernon L. Rooney

Vernon L. Rooney, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 105 to DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

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

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Mr. R. W. Capstick
Vermont Yankee Nuclear Power Corporation

Vermont Yankee Nuclear Power Station

cc:

Mr. J. Gary Weigand
President & Chief Executive Officer
Vermont Yankee Nuclear Power Corp.
R.D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. John DeVincentis, Vice President
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

New England Coalition on Nuclear Pollution
Hill and Dale Farm
R.D. 2, Box 223
Putney, Vermont 05346

Vermont Public Interest Research Group, Inc.
43 State Street
Montpelier, Vermont 05602

William Russell, Regional Administrator
Region I Office
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Mr. R. W. Capstick
Vermont Yankee Nuclear
Power Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

R. K. Gad III
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

W. P. Murphy, Vice President and Manager of Operations
Vermont Yankee Nuclear Power Corp.
R.D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Gerald Tarrant, Commissioner
Vermont Department of Public Service
120 State Street
Montpelier, Vermont 05602

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

G. Dean Weyman
Chairman, Board of Selectman
Post Office Box 116
Vernon, Vermont 05354

Raymond N. McCandless
Vermont Division of Occupational and Radiological Health
Administration Building
Montpelier, Vermont 05602

Honorable John J. Easton
Attorney General
State of Vermont
109 State Street
Montpelier, Vermont 05602

Conner & Wetterhahn, P.C.
Suite 1050
1747 Pennsylvania Avenue, N.W.
Washington, D.C. 20006

cc:

Ellyn R. Weiss, Esq.
Harmon & Weiss
2001 S Street, N.W.
Washington, D.C. 20009

David J. Mullett, Esq.
Special Assistant Attorney General
Vermont Depart. of Public Service
120 State Street
Montpelier, VT 05602

Jay Gutierrez
Regional Counsel
USNRC, Region I
475 Allendale Road
King of Prussia, PA 19406

G. Dana Bisbee, Esq.
Office of the Attorney General
Environmental Protection Bureau
State House Annex
25 Capitol Street
Concord, NH 03301-6397

Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Mr. Glenn O. Bright
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, Vermont 05354

Carol S. Sneider, Esq.
Assistant Attorney General
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Geoffrey M. Huntington, Esquire
Office of the Attorney General
Environmental Protection Bureau
State House Annex
25 Capitol Street
Concord, NH 03301-6397

Charles Bechhoefer, Esq.
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. James H. Carpenter
Administrative Judge
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Adjudicatory File (2)
Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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. 105
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated December 9, 1987, 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 2.C(2) of Facility Operating License No. DPR-28 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 105, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 30 days after the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G McDonald Jr

Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 4, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 105

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove Pages	Insert Pages
39	39
53	53
61	61
65	65
67	67

TABLE 3.2.1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

Automatic Depressurization			
Minimum Number of Operable Instrument Channels per Trip System (Note 4)	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation are Not Satisfied
2	Low-Low Reactor Vessel Water Level	Same as Core Spray	Note 6
2	High Drywell Pressure	≤2.5 psig	Note 6
1	Time Delay (2E-K5A and B)	≤120 seconds	Note 6
1	Bus Power Monitor	--	Note 6
1	Trip System Logic	--	Note 6
2	Time Delay (2E-K16A and B, 2E-K17A and B)	≤8 minutes	Note 6

TABLE 4.2.1 (Continued)

Automatic Depressurization System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
High Drywall Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic (Except Solenoids of Valves)	Every 6 Months (Notes 2 and 11)	Every 6 Months (Note 3)	—

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 excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

5. Check control rod position indication while performing the surveillance requirement of Section 3.3.

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.

3.2 (Continued)

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit. For single recirculation loop operation, the APRM rod block trip setting is reduced in accordance with the analysis presented in NEDE-30060, February 1983. This adjustment accounts for the difference between the single loop and two-loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the fuel cladding integrity safety limit.

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.

4.2 PROTECTIVE INSTRUMENTATION

The protective instrumentation systems covered by this Specification are listed in Table 4.2. Most of these protective systems are composed of two or more independent and redundant subsystems which are combined in a dual-channel arrangement. Each of these subsystems contains an arrangement of electrical relays which operate to initiate the required system protective action.

The relays in a subsystem are actuated by a number of means, including manually-operated switches, process-operated switches (sensors), bistable devices operated by analog sensor signals, timers, limit switches, and other relays. In most cases, final subsystem relay actuation is obtained by satisfying the logic conditions established by a number of these relay contacts in a logic array. When a subsystem is actuated, the final subsystem relay(s) can operate protective equipment, such as valves and pumps, and can perform other protective actions, such as tripping the main turbine generator unit.

With the dual-channel arrangement of these subsystems, the single failure of a relay circuit can be tolerated because the redundant subsystem or system (in the case of high pressure coolant injection) will then initiate the necessary protective action. If a failure in one of these circuits occurs in such a way that an action is taken, the operator is immediately alerted to the failure. If the failure occurs and causes no action, it could then remain undetected, causing a loss of the redundancy in the dual-channel arrangement. Losses in redundancy of this nature are found by periodically testing the relay circuits in the subsystems to ensure that they are operating properly.

It has been the practice in boiling water reactor plants to functionally test protective instrumentation sensors and sensor relays on-line on a monthly frequency. Since logic circuit tests result in the actuation of plant equipment, testing of this nature was done while the plant was shutdown for refueling. In this way, the testing of equipment would not jeopardize plant operation. However, a refueling interval could be as long as eighteen months, which is too long a period to allow an undetected failure to exist.

This Specification is a periodic testing program which is based upon the overall on-line testing of protective instrumentation systems, including logic circuits as well as sensor circuits. Table 4.2 outlined the test, calibration, and logic system functional test schedule for the protective instrumentation systems. The testing of a subsystem includes a functional test of each relay wherever practicable. The testing of each relay includes all circuitry necessary to make the relay operate, and also the proper functioning of the relay contacts. Testing of the automatic initiation inhibit switches verifies the proper operability of the switches and relay contacts. Functional testing of the inaccessible temperature switches associated with the isolation systems is accomplished remotely by application of a heat source to individual switches.

All subsystems are functionally tested, calibrated, and operated in their entirety if practicable. Certain exceptions are necessary because the actuation of certain relays would jeopardize plant operation or present an operational hardship.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 105

TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated December 9, 1987 (W. F. Murphy, VYNPC, to T. Murley, USNRC, FVY 87-117) the Vermont Yankee Nuclear Power Corporation (the licensee) proposed changes to Appendix A of the operating license for the Vermont Yankee Nuclear Power Station dealing with the addition of equipment to the Automatic Depressurization System (ADS) and associated Technical Specifications. The modifications were described and committed to by the licensee in a VYNPC letter dated October 31, 1986. Specifically, the changes included identification of the additional instrumentation in the TS Table 3.2.1 (Emergency Core Cooling System Actuation Instrumentation). The staff has reviewed the submittal and has prepared the following evaluation.

2.0 EVALUATION

2.1 Conformance to Requirements of NUREG-0737, Item II.K.3.18

The revised Technical Specification changes reflect the addition of a bypass timer and a manual inhibit switch to the ADS/low pressure ECCS pump logic. The modifications were made in response to NUREG-0737, Item II.K.3.18 "ADS Logic Modification" and were identified as one of the approved options in the NRC evaluation of the BWR Owners' Group generic response to the subject NUREG-0737 item. The staff safety evaluation was provided to the licensee by NRC letter dated June 3, 1983. The staff safety evaluation stated that the time delay associated with the manual inhibit switch must be justified by analysis and the facility Technical Specifications must be modified to require testing of the timer. The licensee's submittal included discussion of both of these requirements.

The bypass timer setting of eight minutes selected by the licensee is based on avoidance of excessive fuel cladding heatup predicted by analyses using 10 CFR Part 50, Appendix K, Loss of Coolant Accident evaluation models and a demonstration that adequate core cooling is assured for limiting isolation events when the ADS blowdown is delayed 10 minutes after the reactor pressure vessel water level decreases to a low-low level setpoint for a sustained period. The eight minute bypass timer delay is the minimum time interval assumed in generic analyses supporting the proposed modification and has been found acceptable on other applications and is acceptable for Vermont Yankee.

The staff safety evaluation transmitted to the licensee on June 3, 1983 included a requirement for a surveillance test of the bypass timer. The proposed TS revision addresses this requirement and is acceptable.

2.2 Technical Specification Changes

The Vermont Yankee Technical Specification changes resulting from the accepted proposal are as follows:

1) Table 3.2.1, page 39: The Table identifying Emergency Core Cooling Actuation Instrumentation has been revised to add the bypass timer data associated with the ADS inhibit switches.

2) Table 4.2.1, page 53 and Table 4.2 NOTES, page 61: The Tables identifying the testing and surveillance requirements for the Automatic Depressurization System trip system logic have been revised to include new requirements associated with the approved modifications.

3) BASES Sections 3.2/4.2, pages 65 and 67: The BASES sections associated with protective instrumentation are amended to include a discussion of the approved modifications.

The replacement pages identified in the licensee's request are consistent with the supporting analysis and are therefore acceptable as proposed.

The staff has reviewed the basis material provided by the licensee for the proposed TS changes to accommodate the ADS modifications to Vermont Yankee Nuclear Power Plant in response to NUREG-0737, Item II.K.3.18 "ADS Logic Modification." Based on the results of our review discussed in Section 2.0 of this SER, we conclude that the proposed Technical Specification changes are consistent with the design changes and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, an environmental assessment related to this amendment has been published (53 FR 29402) in the Federal Register on August 4, 1988. The Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

4.0 CONCLUSION

We have 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, and (2) 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.

Principal Contributor: M. McCoy

Dated: August 4, 1988