February 24, 1999

Mr. Gregory A. Maret
Director of Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, VT 05301

SUBJECT: ISSUANCE OF AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-28, VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. M97443)

Dear Mr. Maret:

The Commission has issued the enclosed Amendment No. 168 to Facility Operating License No. DPR-28, for the Vermont Yankee Nuclear Power Station in response to your application dated December 10, 1996, as supplemented on January 22, 1999. In your submittal, you proposed to relocate certain fire protection requirements from the Technical Specifications to the Vermont Yankee Fire Protection Plan and Final Safety Analysis Report.

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

Sincerely,

original signed by:
Richard P. Croteau, Project Manager
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No.168 to License No. 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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DATED: February 24, 1999

AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-28 VERMONT YANKEE NUCLEAR POWER STATION

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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. 168
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 December 10, 1996, as supplemented on January 22, 1999, 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:

    (B) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 168 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

The license is also amended by modifying paragraph 3.F to read as follows:

    F. Vermont Yankee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated January 13, 1978, and supplemental SERs, dated 2/20/80, 10/24/80, 1/13/83, 3/25/86, 12/8/89, 6/9/97, 8/12/97, 9/2/98, and 2/24/99 , subject to the following provisions:

    Vermont Yankee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3. This license amendment is effective as of its date of issuance, to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Elinor G. Adensam

Elinor G. Adensam, Director
Project Directorate I-2
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications and Operating License*

Date of Issuance: February 24, 1999

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*Page 7 is attached, for convenience, for the composite license to reflect this change.

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace page 7 of Facility Operating License No. DPR-28 with the attached revised page.

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

Remove	Insert
v	v
1	1
2	2
3	3
4	4
5	5
216	216
240	240
241	241
242	242
243	243
244	244
245	245
246	246
247	247
248	248
249	249
250	250
251	251
252	252
256	256
259	259
262	262
275	275

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

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

F. Vermont Yankee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated January 13, 1978, and supplemental SERs, dated 2/20/80, 10/24/80, 1/13/83, 3/25/86, 12/8/89, 6/9/97, 8/12/97, 9/2/98, and ___________, subject to the following provisions:

Vermont Yankee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

3.G Security Plan

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

A-107
8.25.88
10.20.88

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(Continued)

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Reportable Occurrence - The equivalent of a reportable event which shall be any of the conditions specified in Section 50.73 to 10CFR Part 50.

B. Alteration of the Reactor Core - The act of moving any component affecting reactivity within the reactor vessel in the region above the core support plate, below the upper grid and within the shroud. Normal movement of control rods or neutron detectors, or the replacement of neutron detectors is not defined as a core alteration.

C. Hot Standby - Hot standby means operation with the reactor critical and the main steam line isolation valves closed.

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument including actuation, alarm, or trip. Response time as specified is not part of the routine instrument calibration but will be checked once per operating cycle.

F. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

G. Instrument Functional Test - An instrument functional test shall be:

1. Analog channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

2. Bistable channels - the injection of a signal into the sensor to verify the operability including alarm and/or trip functions.

H. Log System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

1.0 DEFINITIONS

I. Minimum Critical Power Ratio - The minimum critical power ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the appropriate NRC-approved critical power correlation to the actual assembly operating power.

J. Mode - The reactor mode is that which is established by the mode-selector-switch.

K. Operable - A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

L. Operating - Operating means that a system or component is performing its intended functions in its required manner.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment, which are not required to be open during accident conditions, are closed. Such valves may be opened intermittently under administrative controls.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

O. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one

1.0 DEFINITIONS

or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1593 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1593 thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser volume trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

1.0 DEFINITIONS

2. The standby gas treatment system is operable.

3. All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

V. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater that 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

3. Shutdown means conditions as above such that the effective multiplication factor (K_{eff}) of the core shall be less than 0.99.

W. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate circuit in question.

X. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Y. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The operating cycle interval is considered to be 18 months and the tolerance stated above is applicable.

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Deleted

BB. Source Check - The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

CC. Dose Equivalent I-131 - The dose equivalent I-131 shall be that concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion

1.0 DEFINITIONS

factors used for this calculation shall be those listed in NRC Regulatory Guide 1.109, Revision 1, October 1977.

DD. Solidification - Solidification shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements. Suitable forms include dewatered resins and filter sludges.

EE. Deleted

FF. Site Boundary - The site boundary is shown in Figure 2.2-5 in the FSAR.

GG. Deleted

HH. Deleted

II. Off-Site Dose Calculation Manual (ODCM) - A manual containing the current methodology and parameters used in the calculation of off-site doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduction of the environmental radiological monitoring program.

JJ. Process Control Program (PCP) - A process control program shall contain the sampling, analysis, tests, and determinations by which wet radioactive waste from liquid systems is assured to be converted to a form suitable for off-site disposal.

KK. Gaseous Radwaste Treatment System - The Augmented Off-Gas System (AOG) is the gaseous radwaste treatment system which has been designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off-gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

LL. Ventilation Exhaust Treatment System - The Radwaste Building and AOG Building ventilation HEPA filters are ventilation exhaust treatment systems which have been designed and installed to reduce radioactive material in particulate form in gaseous effluents by passing ventilation air through HEPA filters for the purpose of removing radioactive particulates from the gaseous exhaust stream prior to release to the environment. Engineered safety feature atmospheric cleanup systems, such as the Standby Gas Treatment (SBGT) System, are not considered to be ventilation exhaust treatment system components.

MM. Vent/Purging - Vent/Purging is the controlled process of discharging air or gas from the primary containment to control temperature, pressure, humidity, concentration or other operating conditions.

NN. Core Operating Limits Report - The Core Operating Limits Report is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.7.A.4. Plant operation within these operating limits is addressed in individual specifications.

3.10 LIMITING CONDITIONS FOR OPERATION

4.10 SURVEILLANCE REQUIREMENTS

c. From and after the date that one of the two 24 volt ECCS Instrumentation Battery Systems is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding three days unless such Battery System is sooner made operable.

d. From and after the date that the AS-2 125 Volt battery system is made or found to be inoperable for any reason, continued reactor operation is permissible provided Diesel Generator DG-1-1A control power is transferred to Station Battery B1.

e. From and after the date that one of the two 24 Volt Neutron Monitoring and Process Radiation Monitoring battery systems is found or made to be inoperable for any reason, continued reactor operation is permissible providing the minimum channel requirements of Sections 3.1 and 3.2 for the Neutron Monitoring and Process Radiation Monitoring systems are met.

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

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

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

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

f. Investigate reported instances of violations of Technical Specifications, such investigations to include reporting, evaluation, and recommendations to prevent recurrence, to the Manager of Operations.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Nuclear Safety Audit and Review Committee.

h. Review of the Fire Protection Program and implementing procedures, and submittal of recommended changes to the Nuclear Safety Audit and Review Committee.

7. Authority

a. The Plant Operation Review Committee shall be advisory.

b. The Plant Operation Review Committee shall recommend to the Plant Manager approval or disapproval of proposals under Items 6 (a) through (d) above.

1. In the event of disagreement between the recommendations of the Plant Operation Review Committee and the actions contemplated by the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed with immediate notification to the Manager of Operations.

c. The Plant Operation Review Committee shall make tentative determinations as to whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the Nuclear Safety Audit and Review Committee.

8. Records

Minutes shall be kept at the plant of all meetings of the Plant Operation Review Committee and copies shall be sent to the Manager of Operations and the Nuclear Safety Audit and Review Committee.

B. Nuclear Safety Audit and Review Committee

1. The Committee shall consist of at least six (6) persons:

a. Chairman

b. Vice Chairman

c. Four technically qualified persons who are not members of the plant staff.

d. No more than three members shall be selected from the organization reporting to the Manager of Operations.

e. The Committee will obtain advice and counsel from scientific or technical personnel employed by the Company or other organizations whenever the Committee considers it necessary to obtain further scientific or technical assistance in carrying out its responsibilities.

Any reportable occurrence shall be reported to the Manager of Operations and shall be reviewed by the Plant Operations Review Committee. This Committee shall prepare a separate, sequentially numbered, report for each reportable occurrence. Each report shall describe the circumstances leading up to and resulting from the occurrence, the corrective action taken by the shift, an attempt to define the cause of the occurrence, and shall recommend appropriate action to prevent or reduce the probability of a repetition of the occurrence.

Copies of all such reports shall be submitted to the Chairman of the Nuclear Safety Audit and Review Committee for review and to the Manager of Operations for review and approval of any recommendations.

6.4 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Applies to administrative action to be followed in the event a safety limit is exceeded.

If a safety limit is exceeded, the reactor shall be shutdown immediately. An immediate report shall be made to the Manager of Operations. A complete analysis of the circumstances leading up to and resulting from the situation together with recommendations by the Plant Operations Review Committee shall also be prepared. This report shall be submitted to the Manager of Operations and the Chairman of the Nuclear Safety Audit and Review Committee.

Reactor operation shall not be resumed until authorized by the U.S. Nuclear Regulatory Commission.

6.5 PLANT OPERATING PROCEDURES

A. Detailed written procedures, involving both nuclear and non-nuclear safety, including applicable check-off lists and instructions, covering areas listed below shall be prepared and approved.

All procedures shall be adhered to.

1. Normal startup, operation and shutdown of systems and components of the facility.
2. Refueling operations.
3. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, suspected Primary System leaks and abnormal reactivity changes.
4. Emergency conditions involving potential or actual release of radioactivity.
5. Preventive and corrective maintenance operations which could have an effect on the safety of the reactor.
6. Surveillance and testing requirements.
7. Fire protection program implementation.

e. Land Use Census, Specification 3.9.D

With a land use census not being conducted as required by Specification 3.9.D, prepare and submit to the Commission within 30 days a special report which identifies the reasons why the survey was not conducted, and what steps are being taken to correct the situation.

3. Environmental Radiological Monitoring

Radiological Environmental Surveillance Reports covering the operation of the unit during previous calendar year shall be submitted prior to May 1 of each year.

The annual Radiological Environmental Surveillance Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impact of the plant operation on the environment.

The annual Radiological Environmental Surveillance Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period pursuant to the table and figures in the ODCM. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

With the level of radioactivity in an environmental sampling media at one or more of the locations specified in Table 3.9.3 exceeding the reporting levels of Table 3.9.4, the condition shall be described in the next annual Radiological Environmental Surveillance Report only if the measured level of radioactivity was not the result of plant effluents. With the radiological environmental monitoring program not being conducted as specified in Table 3.9.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence shall be included in the next annual Radiological Environmental Surveillance Report.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.168 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

The Vermont Yankee Nuclear Power Station is a boiling water reactor (BWR), model BWR-4, with a Mark I containment. By letter dated December 10, 1996, as supplemented on January 22, 1999, the Vermont Yankee Nuclear Power Corporation, the licensee for the Vermont Yankee Nuclear Power Station, submitted for Nuclear Regulatory Commission (NRC) staff review a proposed change to the Technical Specifications (TS). The licensee proposed to relocate certain fire protection requirements to the Vermont Yankee Fire Protection Plan and Final Safety Analysis Report (FSAR). The requested changes are consistent with the guidance of NRC Generic Letters (GL) 86-10, Implementation of Fire Protection Requirements, and 88-12, Removal of Fire Protection Requirements from the Technical Specifications. The January 22, 1999, submittal provided supplemental updated information that did not expand the scope of the amendment request as originally noticed or require a change to the original no significant hazards determination.

Specifically, the licensee proposes to replace the current license condition Section 3.F with the license condition recommended by GL 86-10. In addition, the licensee proposes to relocate certain current fire protection TS requirements to the Vermont Yankee Fire Protection Plan and FSAR. Finally, the licensee proposes new TS 5.2.A.6.h, which states clearly that the responsibilities of the Plant Operations Review Committee include the review of the Fire Protection Program and implementing procedures, and the submittal of recommended changes to the Nuclear Safety Audit and Review Committee.

2.0 EVALUATION

2.1 Background

Section 182a of the Atomic Energy Act of 1954, as amended (the Act), requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in Title 10 of the Code of Federal Regulations (10 CFR), Section 50.36. That regulation requires that the TS include items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety." The criteria set forth in the policy statement have been incorporated into 10 CFR 50.36 (60 FR 36953).

Section 50.48, "Fire protection," of 10 CFR Part 50 requires that each operating nuclear power plant have a fire protection plan that satisfies Criterion 3, "Fire protection," of 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants. The fire protection plan must describe the overall fire protection program for the facility; outline the plans for fire protection, fire detection, and fire suppression capability; and limitations of fire damage. The program must also describe specific features necessary to implement the program, such as administrative controls and personnel requirements for fire prevention and manual fire suppression activities; automatic and manually operated fire detection and suppression systems; and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured. The NRC staff initially approved the Vermont Yankee Nuclear Power Station fire protection program in its fire protection Safety Evaluation dated January 13, 1978.

Following the fire at the Browns Ferry Nuclear Power Plant on March 22, 1975, the Commission undertook a number of actions to ensure that improvements were implemented in the fire protection programs for all power reactor facilities. Because of the extensive modification of fire protection programs and the number of open issues resulting from staff evaluations, a number of revisions and alterations occurred in these programs over the years. Consequently, licensees were requested by GL 86-10 to incorporate the final NRC-approved fire protection program in their FSARs. In this manner, the fire protection program, including the systems, certain administrative and technical controls, the organization, and other plant features associated with fire protection, would have a status consistent with that of other plant features described in the FSAR. In addition, the Commission concluded that a standard license condition, requiring compliance with the provisions of the fire protection program as described in the FSAR, should be used to ensure uniform enforcement of the fire protection requirements. Finally, the Commission stated that with the required actions, licensees may request an amendment to delete the fire protection TS that would now be unnecessary. Subsequently, the NRC issued GL 88-12 to give guidance for the preparation of a license amendment request to implement GL 86-10.

GL 86-10 and GL 88-12 refer to removing fire protection requirements from the TS. License amendments that relocate the fire protection requirements in accordance with GL 86-10 and GL 88-12 do not revise the requirements for fire protection operability, testing, or inspections. Such amendments simply replace the fire protection TS sections with the standard fire protection license condition. The license condition implements and maintains the NRC-approved fire protection program, including the fire protection requirements previously specified in the TS, in accordance with 10 CFR 50.48. Therefore, such amendments, including the one proposed by the licensee, are administrative in nature and have no effect on public health and safety.

2.2 Technical Evaluation

The specific TS changes proposed by the licensee are:

1. Revise the current License Condition 3.F addressing fire protection as follows:

   F. Vermont Yankee shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report for the facility and as approved in the SER dated January 13, 1978, and supplemental SERs, dated 2/20/80, 10/24/80, 1/13/83, 3/25/86, 12/8/89, 6/9/97, 8/12/97, 9/2/98, and 2/24/99, subject to the following provisions:

   Vermont Yankee may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

2. Delete the following TS and relocate the requirements to the Vermont Yankee Fire Protection Plan:

   a. TS Definition AA, Vital Fire Water Suppression System

   b. TS 3.10, fire watch requirements for the cable vault area

   c. TS 3.13/4.13, Fire Protection

   d. TS 6.1.E and page note

   e. TS 6.5.A.7, fire protection program implementation details

   f. TS 6.7.C.2.f, Vitial Fire Protection System

3. Insert the following TS Section 6.2.A.6.h to require that the responsibilities of the Plant Operations Review Committee include the review of the Fire Protection program and the implementing procedures, and the submittal of recommended changes to the Nuclear Safety Audit and Review Committee.

The NRC staff reviewed the license amendment request against the guidance provided in GLs 86-10 and 88-12. Generic Letter 86-10 requests that licensees incorporate the NRC-approved fire protection program in the plant's FSAR and recommends a standard fire protection license condition. The amendment would revise the current fire protection license condition to be consistent with the standard fire protection license condition recommended in GL 86-10.

Generic Letter 88-12 addresses the elements that licensees should include in a license amendment request to remove fire protection requirements from their TS. These elements are (1) that the NRC-approved fire protection program must be incorporated into the FSAR; (2) the limiting conditions for operation and surveillance requirements associated with fire detection systems, fire suppression systems, fire barriers, and the administrative controls that address fire brigade staffing, would be relocated from the TS (the existing administrative controls associated with fire protection audits and specifications related to the capability for safe shutdown following a fire would be retained); (3) all operational conditions, remedial actions, and test requirements presently included in the TS for these systems, as well as fire brigade staffing requirements, shall be incorporated into the fire protection program; (4) the standard fire protection license condition specified in GL 86-10 must be included in the facility operating license; (5) the Onsite Review Group shall be given responsibility for the review of the fire protection program and implementing procedures, and for the submittal of recommended changes to the Company Nuclear Review and Audit Group (Offsite or Corporate Review Group); and (6) the Administrative Controls section of the TS shall provide that fire protection program implementation shall be one of the elements for which written procedures shall be established, implemented, and maintained.

Vermont Yankee has committed to incorporate the NRC-approved fire protection program requirements into the FSAR. By letter dated January 22, 1998, the licensee affirmed that the requirements have been duplicated in the Technical Requirements Manual (TRM) and the TRM has been incorporated by reference into the FSAR. The licensee has, therefore, satisfied Elements 1, 2, and 3 of GL 88-12.

This amendment would revise the current fire protection license condition to be consistent with the standard fire protection license condition recommended in GL 86-10. The licensee will, therefore, satisfy Element 4 of GL 88-12 with issuance of the proposed amendment.

To satisfy Element 5 of GL 88-12, the licensee proposed adding TS 6.2.A.6.h, described previously. The licensee will, therefore, satisfy Element 5 of GL 88-12 with issuance of the proposed amendment.

Element 6 of GL 88-12 specifies that the fire protection program implementation shall be one of the elements for which written procedures shall be established, implemented, and maintained. Section 6.5.A.7 of the current TS requires that written procedures be established, implemented, and maintained for implementing the fire protection program. This TS will remain in place following issuance of the proposed amendment. Thus, no changes are required and the licensee has, therefore, satisfied Element 6 of GL 88-12.

The licensee's proposed TS amendments are in accordance with NRC staff guidance provided in GL 86-10 and GL 88-12.

In summary, the licensee has proposed to incorporate the existing TS fire protection requirements as previously stated into its fire protection program, TRM, and FSAR. The fire protection program and TRM are, by reference, incorporated in the FSAR. This conforms to staff guidance in GL 86-10, "Implementation of Fire Protection Requirements," and GL 88-12, "Removal of Fire Protection Requirements from Technical Specifications," for removing unnecessary fire protection TS in four major areas: fire detection systems, fire suppression systems, fire barriers, and fire brigade staffing requirements. In addition, incorporating these requirements in the FSAR is consistent with NUREG-1433, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/4)" and 10 CFR 50.36, as amended, because these requirements do not impact reactor operations, do not identify a parameter that is an initial condition assumption for a design-basis accident or transient, do not identify a significant abnormal degradation of the reactor coolant pressure boundary, and do not provide any mitigation of a design-basis event.

The fire protection plan required by 10 CFR 50.48, as implemented and maintained by the fire protection license condition, provides reasonable assurance that fires will not give rise to an immediate threat to public health and safety. Although there are aspects of fire detection and mitigation functions that have been determined to be risk-significant, such that Criterion 4 of 10 CFR 50.36 would otherwise seem to apply, the minimum requirements for those functions were established in General Design Criterion 3 and 10 CFR 50.48, and further controls are not necessary, because the licensee must comply with these minimum requirements regardless of whether they are restated in the TS or not.

The licensee's fire protection program is required by 10 CFR 50.48, and any changes to the program are governed by 10 CFR 50.48 and License Condition 3.F, as previously described. Therefore, the requirements relocated to the licensee's fire protection program may be controlled in accordance with 10 CFR 50.59.

2.3 Conclusion

These relocated requirements relating to fire protection features are not required to be in the TS under 10 CFR 50.36 or other regulations, or by Section 182a of the Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.48 and 10 CFR 50.59 to address future changes to these requirements. The licensee's proposed TS amendments are in accordance with NRC staff guidance provided in GL 86-10 and GL 88-12. Accordingly, the staff has concluded that the proposed TS changes are 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 affecting issuance of this TS amendment.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes administrative requirements, or 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 (62 FR 8801, February 26, 1997). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). 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.

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