Docket Files

Docket No. 50-271

July 21, 1989

Mr. R. W. Capstick
Licensing Engineering
Vermont Yankee Nuclear Power Corporation
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Capstick:

SUBJECT: ISSUANCE OF AMENDMENT NO. 114 TO DPR-28 - VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. 66873)

The Commission has issued the enclosed Amendment No. 114 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated December 7, 1987. Clarifying information was provided July 15, 1988 and June 8, 1989.

This amendment modifies the Technical Specifications to eliminate the present requirements to test the remaining train(s) of the ECCS and SLC systems immediately and daily thereafter when one train has a component out of service. The systems will be deemed operable based on testing within 24 hours and on a monthly basis thereafter.

A copy of our Safety Evaluation is also enclosed. A Notice of Issuance has been forwarded to the Office of the Federal Register for publication.

This completes action under TAC 66873.

Sincerely

Original Signed By

Morton B. Fairtile, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 114 to
   License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

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1/1

VY AMENDMENT TAC 66873

*See previous concurrence

cp-1

OFC: PDI-3*	:PDI-3*	:DIR/PDI-3*	:OGC*	:	:	:
NAME:MRushbrook	:MFairtile	:RWessman	:R.Bachmann	:	:	:
DATE:7/10/89	:7/10/89	:7/18/89	:7/03/89	:	:	

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Adjudicatory File (2)
Atomic Safety and Licensing Board
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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. 114
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated December 7, 1987 as clarified on July 15, 1988, and June 8, 1989, 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 set forth in 10 CFR Chapter I;

   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:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 114 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by

Richard H. Wessman, 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: July 21, 1989

PDI-3/PM MBF	PDI-3/LA	OGC	PDI-3/D
MBFairtile	MRushbrook	RBachmann	RHWessman
7/10/89	/ /89	7/13/89	7/18/89

ATTACHMENT TO LICENSE AMENDMENT NO. 114

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
80	80
83	83
86	86
87	87
88	88
89	89
90	90
91	91
92R	92R
93	93
94	94
99	99
100	100
103	103
104	104
131	131
-	131a*
176	176
177	177

*Denotes new page

3.4 LIMITING CONDITIONS FOR OPERATION

B. <u>Operation with Inoperable Components</u>

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

C. <u>Liquid Poison Tank - Boron Concentration</u>

At all times when the Standby Liquid Control System is required to be operable, the following conditions shall be met:

1. The net volume versus concentration of the sodium pentaborate solution in the standby liquid control tank shall meet the requirements of Figure 3.4.1.

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

B. <u>Operation with Inoperable Components</u>

When a component becomes inoperable, its redundant component shall be or shall have been demonstrated to be operable within 24 hours.

C. <u>Liquid Poison Tank - Boron Concentration</u>

1. The solution volume in the tank and temperature in the tank and suction piping shall be checked at least daily.

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

Bases:

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

A. Normal Operation

The design objective of the Reactor Standby Liquid Control System is to provide the capability of bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. To meet this objective, the Liquid Control System is designed to inject a quantity of boron which produces a concentration of 800 ppm of boron in the reactor core in less than 138 minutes. An 800 ppm boron concentration in the reactor core is required to bring the reactor from full power to a 5% Δk subcritical condition. An additional margin (25% of boron) is added for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3850 gallons of solution having a 10.1% sodium pentaborate concentration is required to meet this shutdown requirement.

The time requirement (138 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak. For a required minimum pumping rate of 35 gallons per minute, the maximum net storage volume of the boron solution is established as 4830 gallons.

Boron concentration, solution temperature, and volume are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Experience with pump operability indicates that testing at monthly intervals is adequate to detect if failures have occurred. Flow, relief valve, circuitry, and trigger assembly testing at the prescribed intervals assures a high reliability of system operation capability. Recirculation of the borated solution is done during each operating cycle to ensure one suction line from the boron tank is clear.

B. Operation With Inoperable Components

Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. Assurance that the system will perform its intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements. Whenever one redundant component is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant component be tested within 24 hours.

C. Liquid Poison Tank - Boron Concentration

The solution saturation temperature varies with the concentration of sodium pentaborate. The solution shall be kept at least 10°F above the saturation temperature to guard against boron precipitation. The 10°F margin is included in Figure 3.8.3 of the FSAR. Temperature and liquid level alarms for the system are annunciated in the Control Room.

3.5 LIMITING CONDITION FOR OPERATION

2. From and after the date that one of the Core Spray Subsystems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days, all active components of the other Core Spray Subsystem, the LPCI Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

3. From and after the date that one of the LPCI pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such pump is sooner made operable, provided that during such seven days, the remaining active components of the LPCI Containment Cooling Subsystem and all active components of both Core Spray Subsystems and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

For the purpose of performing inspection/repair of the impeller wear rings on each LPCI pump during the 1986-1987 operating cycle, reactor operation is permissible only during the succeeding 14 days from the time each pump is made inoperable unless such pump is sooner made operable. Additionally, during the out-of-service time, the remaining

4.5 SURVEILLANCE REQUIREMENT

Item		Frequency
c.	Pump and Motor-Operated Valve Operability except Recirculation Pump discharge valves.	Once/month

2. When one Core Spray Subsystem is made or found to be inoperable, the active components of the redundant Core Spray Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3. When one of the LPCI pumps is made or found to be inoperable, the remaining operable LPCI pumps shall have been or shall be demonstrated to be operable within 24 hours.

During the inspection/repair of the impeller wear rings on each LPCI pump during the 1986-1987 operating cycle, the remaining active components of the LPCI Subsystem (except Recirculation System discharge valves), the Containment Cooling Subsystems, both Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall continue to be demonstrated operable on a monthly basis as specified in Technical Specifications 4.5.A.1 and 4.10.A.1.

3.5 LIMITING CONDITION FOR OPERATION

active components of the LPCI Containment Cooling Subsystem and all active components of both Core Spray Subsystems and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during such seven days all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the diesel generators required for operation of such components if no external source of power were available, shall be operable.

5. All recirculation pump discharge valves and bypass valves shall be operable or closed prior to reactor startup.

6. If the requirements of Specification 3.5.A cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in a cold shutdown condition with 24 hours.

B. Containment Spray Cooling Capability

1. Both containment cooling spray loops are required to be operable when the reactor water temperature is greater than 212°F except that a Containment

4.5 SURVEILLANCE REQUIREMENT

4. When a LPCI Subsystem is made or found to be inoperable, the active components of the redundant LPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours (except the Recirculation System discharge valves).

5. a. All recirculation pump discharge and bypass valves shall be tested for operability during any period of reactor cold shutdown exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

   b. Recirculation Pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds each refueling outage.

B. Containment Spray Cooling Capability

1. Surveillance of the drywell spray loops shall be performed as follows. During each five-year period, an air test shall be performed on the drywell spray headers and nozzles.

3.5 LIMITING CONDITION FOR OPERATION

Cooling Subsystem may be inoperable for thirty days.

2. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in Specifications 3.5.C.2 and 3.5.C.3 below, both RHR Service Water Subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that one of the RHR service water pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR Service Water Subsystem are operable.

4.5 SURVEILLANCE REQUIREMENT

2. When a Containment Cooling Subsystem is made or found to be inoperable, the active components of the redundant Containment Cooling Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

C. Residual Heat Removal (RHR) Service Water System

Surveillance of the RHR Service Water System shall be performed as follows:

1. RHR Service Water Subsystem testing:

   a. Pump and motor-operated valve operability shall be tested every three months.

   b. Each RHR service water pump shall be tested after pump maintenance and every three months. Each pump shall deliver at least 2700 gpm and a pressure of at least 70 psia shall be maintained at the RHR heat exchanger service water outlet when the corresponding pairs of RHR service water pumps and station service water pumps are operating.

2. When one of the RHR service water pumps is made or found to be inoperable, the operable RHR service water pumps shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

3. From and after the date that one RHR Service Water Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that all active components of the other RHR Service Water Subsystem, both Core Spray Subsystems, and both diesel generators required for operation of such components if no external source of power were available, shall be operable.

4. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

D. <u>Station Service Water and Alternate Cooling Tower Systems</u>

1. Except as specified in Specifications 3.5.D.2 and 3.5.D.3, both Station Service Water Subsystem loops and the alternate cooling tower shall be operable

4.5 SURVEILLANCE REQUIREMENT

3. When one RHR Service Water Subsystem is made or found to be inoperable, the active components of the redundant RHR Service Water Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

D. <u>Station Service Water and Alternate Cooling Tower Systems</u>

Surveillance of the Station Service Water and Alternate Cooling Tower Systems shall be performed as follows:

1. Pump and motor-operated valve operability shall be tested every six months and whenever the plant is shutdown, but

3.5 LIMITING CONDITION FOR OPERATION

whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the Station Service Water Subsystems is made or found inoperable for any reason, reactor operation is permissible only during the succeeding 15 days unless such subsystem is made operable, provided that during such 15 days all other active components of the Station Service Water and Alternate Cooling Tower Systems are operable.

3. From and after the date that the Alternate Cooling Tower Subsystem or both Station Service Water Subsystems are made or found inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystems are made operable, provided that during such seven days all other active components of the other subsystem are operable.

4. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

not more than every three months. Flow rate test each station service water pump after pump maintenance and every three months. Each pump shall deliver at least 2700 gpm against a TDH of 250 feet.

2. When one Station Service Water Subsystem is made or found to be inoperable, the active components of the redundant Station Service Water Subsystem and the alternate cooling tower fan shall have been or shall be demonstrated to be operable within 24 hours.

3. When the Alternate Cooling Subsystem or both Station Service Water Subsystems are made or found to be inoperable, the operable subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

E. High Pressure Coolant Injection (HPCI) System

1. Except as specified in Specification 3.5.E.2, whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 150 psig and prior to reactor startup from a cold condition:

    a. The HPCI System shall be operable.

    b. The condensate storage tank shall contain at least 75,000 gallons of condensate water.

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components of the Automatic Depressurization Subsystems, the Core Spray Subsystems, the LPCI Subsystems, and the RCIC System are operable.

3. If the requirements of Specification 3.5.E cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

4.5 SURVEILLANCE REQUIREMENT

E. High Pressure Coolant Injection (HPCI) System

Surveillance of HPCI Systems shall be performed as follows:

1. Testing:

Item	Frequency
Simulated Automatic Actuation Test	Each refueling outage
Pump operability	Once/month
Motor-operated valve operability	Once/month
Flow rate test (recirculate to condensate storage tank). The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure.	Once/ operating cycle

2. When the HPCI Subsystem is made or found to be inoperable, the Automatic Depressurization System shall have been or shall be demonstrated to be operable within 24 hours.

    NOTE: Automatic Depressurization System operability shall be demonstrated by performing a functional test of the trip system logic.

3.5 LIMITING CONDITION FOR OPERATION

F. Automatic Depressurization System

1. Except as specified in Specification 3.5.F.2 below, the entire Automatic Depressurization Relief System shall be operable at any time the reactor pressure is above 100 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that one of the four relief valves of the Automatic Depressurization Subsystem are made or found to be inoperable due to malfunction of the electrical portion of the valve when the reactor is pressurized above 100 psig with irradiated fuel in the reactor vessel, continued reactor operation is permissible only during the succeeding seven days unless such a valve is sooner made operable, provided that during such seven days both the remaining Automatic Relief System valves and the HPCI System are operable.

4.5 SURVEILLANCE REQUIREMENT

F. Automatic Depressurization System

Surveillance of the Automatic Depressurization System shall be performed as follows:

1. During each operating cycle each relief valve shall be manually opened with the reactor at low pressure until the thermocouples downstream of the valve indicates fluid is flowing from the valve.

2. When one relief valve of the Automatic Pressure Relief Subsystem is made or found to be inoperable, the HPCI Subsystem shall have been or shall be demonstrated to be operable within 24 hours.

3.5 LIMITING CONDITION FOR OPERATION

3. If the requirements of Specification 3.5.F cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 100 psig within 24 hours.

G. Reactor Core Isolation Cooling System (RCIC)

1. Except as specified in Specification 3.5.G.2 below, the RCIC System shall be operable whenever the reactor pressure is greater than 150 psig and irradiated fuel is in the reactor vessel.

2. From and after the date that the RCIC System is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the HPCI System are operable.

4.5 SURVEILLANCE REQUIREMENT

G. Reactor Core Isolation Cooling System (RCIC)

Surveillance of the RCIC System shall be performed as follows:

1. Testing

Item	Frequency
Pump operability	Once/month
Motor-operated valve operability	Once/month
Flow rate test (recirculate to condensate storage tank). The RCIC shall deliver at least 400 gpm at normal operating pressure.	After major pump maintenance and every three months
Simulated automatic actuation test (testing valve operability)	Each refueling outage

3.5 LIMITING CONDITIONS FOR OPERATION

3. If the requirements of Specification 3.5.G cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 120 psig within 24 hours.

H. Minimum Core and Containment Cooling System Availability

1. During any period when one of the standby diesel generators is inoperable, continued reactor operation is permissible only during the succeeding seven days, provided that all of the Low Pressure Core Cooling and Containment Cooling Subsystems connecting to the operable diesel generator shall be operable. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

2. Any combination of inoperable components in the Core and Containment Cooling Systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

3. When irradiated fuel is in the reactor vessel and the reactor is in the cold shutdown condition, all Core and Containment Cooling Subsystems may be inoperable provided no work is permitted which has the potential for draining the reactor vessel.

4.5 SURVEILLANCE REQUIREMENTS

H. Minimum Core and Containment Cooling System Availability

1. When one of the standby diesel generators is made or found to be inoperable, the remaining diesel generator shall have been or shall be demonstrated to be operable within 24 hours.

Bases:

3.5 CORE AND CONTAINMENT COOLANT SYSTEMS

A. Core Spray Cooling System and Low Pressure Coolant Injection System

This Specification assures that adequate standby cooling capability is available whenever irradiated fuel is in the Reactor Vessel.

Based on the loss-of-coolant analyses, the Core Spray and LPCI Systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit the accident-caused core conditions as specified in 10CFR50, Appendix K. The analyses consider appropriate combinations of the two Core Spray Subsystems and the two LPCI Subsystems associated with various break locations and equipment availability in accordance with required single failure assumptions. (Each LPCI Subsystem consists of the LPCI pumps, the recirculation pump discharge valve, and the LPCI injection valve which combine to inject torus water into a recirculation loop.)

The LPCI System is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system is completely independent of the Core Spray System; however, it does function in combination with the Core Spray System to prevent excessive fuel clad temperature. The LPCI and the Core Spray Systems provide adequate cooling for break areas up to and including the double-ended recirculation line break without assistance from the high pressure emergency Core Cooling Subsystems.

The intent of these specifications is to prevent startup from the cold condition without all associated equipment being operable. However, during operation, certain components may be out of service for the specified allowable repair times. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements. Whenever one redundant system is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant subsystem be tested within 24 hours.

3.5 (Cont'd)

B. and C. Containment Spray Cooling Capability and RHR Service Water System

The containment heat removal portion of the RHR System is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 5 psig and, therefore, the flow is more than ample to provide the required heat removal capability. Reference: Section 14.6.3.3.2 FSAR.

Each Containment Cooling Subsystem consists of two RHR service water pumps, 1 heat exchanger, and 2 RHR (LPCI) pumps. Either set of equipment is capable of performing the containment cooling function. In fact, an analysis in Section 14.6 of the FSAR shows that one subsystem consisting of 1 RHR service water pump, 1 heat exchanger, and 1 RHR pump has sufficient capacity to perform the cooling function. Assurance that the systems will perform their intended function is obtained from the results of the pump and valve testing performed in accordance with ASME Section XI requirements. Whenever one redundant system is inoperable, the potential for extended operation with two subsystems inoperable is reduced by requiring that the redundant subsystem be tested within 24 hours.

D. Station Service Water and Alternate Cooling Tower Systems

The Station Service Water Subsystems and the Alternate Cooling Tower System provide alternate heat sinks to dissipate residual heat after a shutdown or accident. Each Station Service Water Subsystem and the Alternate Cooling Tower System provides sufficient heat sink capacity to perform the required heat dissipation. The Alternate Cooling Tower System will provide the necessary heat sink in the event both Station Service Water Subsystems become incapacitated due to a loss of the Vernon Dam with subsequent loss of the Vernon Pond.

E. High Pressure Coolant Injection System

The High Pressure Coolant Injection System (HPCIs) is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or Core Spray Cooling Subsystems can protect the core.

The HPCIs meets this requirement without the use of outside power. For the pipe breaks for which the HPCIs is intended to function, the core never uncovers and is continuously cooled; thus, no clad damage occurs and clad temperatures remain near normal throughout the transient. Reference: Subsection 6.5.2.2 of the FSAR.

F. Automatic Depressurization System

The relief valves of the Automatic Depressurization System are a backup to the HPCIs. They enable the Core Spray Cooling System or LPCI Subsystem to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the Core Sprays or LPCI Subsystem. Either of the two Core Spray Cooling Systems or LPCIs provides sufficient flow of coolant to prevent clad melting. All four relief valves are included in the Automatic Pressure Relief System. (See VYNPS, FSAR Vol. 4, Appendix B.)

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

A. Core Spray and LPCI

During normal plant operation, manual tests of operable pumps and valves shall be conducted monthly to demonstrate operability with the exception of the Recirculation Pump Discharge valves. The Recirculation System discharge valves are not tested during plant operation since to do so would create a severe plant transient.

During each refueling shutdown, tests (as summarized below) shall be conducted to demonstrate proper automatic operation and system performance.

Periodic testing at the intervals specified above will demonstrate that all components which do not operate during normal conditions will operate properly if required.

The automatic actuation test will be performed by simulation of high drywell pressure or low-low water level. The starting of the pump and actuation of valves will be checked. The normal power supply will be used during the test. Testing of the sequencing of the pumps when the diesel generator is the source of power will be checked during the testing of the diesel. Following the automatic actuation test, the flow rate will be checked by recirculation to the suppression chamber. The pump and valve operability checks will be performed by manually starting the pump or activating the valve. For the pumps, the pump motors will be run long enough for them to reach operating temperatures.

B. and C. Containment Spray Cooling Capability and RHR Service Water Systems

The periodic testing intervals specified in Specifications 4.5.B. and C. will demonstrate that all components will operate properly if required. Since this is a manually actuated system, no automatic actuation test is required. The system will be activated manually and the flow checked by an indicator in the control room.

Once every five years air tests will be performed to assure that the containment spray header nozzles are operable.

4.5 CORE AND CONTAINMENT COOLANT SYSTEMS (Continued)

D., E., and F. Station Service Water and Alternate Cooling Tower Systems and High Pressure Coolant Injection and Automatic Depressurization System

The testing intervals for the HPCI System will demonstrate that the system will operate if required. The Automatic Depressurization System is tested during refueling outages to avoid an undesirable blowdown of the Reactor Coolant System.

The HPCI Automatic Actuation Test will be performed by simulation of the accident signal. The test is normally performed in conjunction with the automatic actuation of all Core Standby Cooling Systems.

A flow rate test of HPCIs is performed once/operating cycle during normal station operation by pumping water at rated conditions from the condensate storage tank and back through the full flow test return line to the tank.

The pump operability check will be performed by starting the turbine manually, valves will also be stroked by manual actuation of the operators.

G. Reactor Core Isolation Cooling System

Frequency of testing of the RCIC System is the same as the HPCIs except the flow rate test is performed after major pump maintenance and every three months, and demonstrates that the system is operable if needed.

H. Minimum Core and Containment Cooling System Availability

Assurance that the diesels will perform their intended function is obtained by the periodic surveillance test and the results obtained from the pump and valve testing performed in accordance with ASME Section XI requirements. Whenever a diesel is inoperable, the potential for extended operation with two diesels inoperable is reduced by requiring that the redundant diesel be tested within 24 hours.

I. Maintenance of Filled Discharge Pipe

Observation of water flowing from the discharge line high point vent monthly assures that the Core Cooling Subsystems will not experience water hammer damage when any of the pumps are started. Core Spray Subsystems and LPCI Subsystems will also be vented through the discharge line high point vent following a return from an inoperable status to assure that the system is "solid" and ready for operation.

3.7 LIMITING CONDITIONS FOR OPERATION

4. If this condition cannot be met, procedures shall be initiated immediately to establish the conditions listed in Specifications 3.7.C.1(a) through (d), and compliance shall be completed within 24 hours thereafter.

C. Secondary Containment System

1. Integrity of the Secondary Containment System shall be maintained during all modes of plant operation except when all of the following conditions are met:

4.7 SURVEILLANCE REQUIREMENTS

f. DOP and halogenated hydrocarbon test shall be performed following any design modification to the Standby Gas Treatment System housing that could have an effect on the filter efficiency.

g. An air distribution test demonstrating uniformity within ±20% across the HEPA filters and charcoal adsorbers shall be performed if the SBGTS housing is modified such that air distribution could be affected.

3. a. At least once per operating cycle automatic initiation of each branch of the Standby Gas Treatment System shall be demonstrated.

b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one circuit of the Standby Gas Treatment System is made or found to be inoperable, the other circuit shall have been or shall be demonstrated to be operable within 24 hours.

C. Secondary Containment System

1. Surveillance of secondary containment shall be performed as follows:

3.7 LIMITING CONDITIONS FOR OPERATION

a. The reactor is subcritical and Specification 3.3.A is met and

4.7 SURVEILLANCE REQUIREMENTS

a. A preoperational secondary containment capability test shall be conducted after isolating the Reactor Building and placing either Standby Gas Treatment System filter train in operation. Such tests shall demonstrate the capability to maintain a 0.15 inch of water vacuum under calm wind ($2 < u < 5$ mph) condition with a filter train flow rate of not more than 1500 cfm.

3.10 LIMITING CONDITIONS FOR OPERATION

B. Operation With Inoperable Components

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in the Cold Condition, the requirements of 3.10.A shall be met except:

1. Diesel Generators

From and after the date that one of the diesel generators or its associated buses are made or found to be inoperable for any reason and the remaining diesel generator is operable, the requirements of Specification 3.5.H.1 shall be satisfied.

2. Batteries

a. From and after the date that ventilation is lost in the Battery Room portable ventilation equipment shall be provided.

b. From and after the date that one of the two 125 volt Station Battery Systems is made or found to be inoperable for any reasons, continued reactor operation is permissible only during the succeeding three days provided Specification 3.5.H is met unless such Battery System is sooner made operable.

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.

4.10 SURVEILLANCE REQUIREMENTS

B. Operation With Inoperable Components

1. Diesel Generator

When one of the diesel generators is made or found to be inoperable, the requirements of Specification 4.5.H.1 shall be satisfied.

2. Batteries

Samples of the Battery Room atmosphere shall be taken daily for hydrogen concentration determination.

3.10 LIMITING CONDITIONS FOR OPERATION

3. Off-Site Power

   a. From and after the date that both startup transformers and one diesel generator or associated buses are made or found to be inoperable for any reason, reactor operation may continue provided the requirements of Specification 3.5.H.1 are satisfied.

   b. From and after the date that both delayed access off-site power sources become unavailable, reactor operation may continue for seven days provided both emergency diesel generators, associated buses, and all Low Pressure Core and Containment Cooling Systems are operable.

4. 480 V Uninterruptible Power Systems

   From and after the date that one Uninterruptible Power System or its associated Motor Control Center are made or found to be inoperable for any reason, the requirements of Specification 3.5.A.4 shall be satisfied.

5. RPS Power Protection

   From and after the date that one of the two redudant RPS power protection panels on an in-service RPS MG set or alternate power supply is made or found to be inoperable, the associated RPS MG set or alternate supply will be taken out of service until the panel is restored to operable status.

4.10 SURVEILLANCE REQUIREMENTS

3. Off-Site Power

   a. When one of the diesel generators or associated buses is made or found to be inoperable, the requirements of Specification 4.5.H.1 shall be satisfied.

   b. When both delayed access off-site power sources are unavailable, both diesel generators and associated buses shall have been or shall be demonstrated to be operable within 24 hours.

4. 480 V Uninterruptible Power Systems

   When it is determined that one Uninterruptible Power System or its associated Motor Control Center is inoperable, the requirements of Specification 4.5.A.4 shall be satisfied.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 114 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 7, 1987, the Vermont Yankee Nuclear Power Corporation, the licensee for the Vermont Yankee Nuclear Power Station, submitted information on the alternate testing requirements of the emergency core cooling system (ECCS) and the standby liquid control (SLC) system and requested that these requirements be eliminated (Refs. 1, 2 and 6). The ECCS includes the diesel generators, the automatic depressurization systems, the high and low pressure core injection systems, containment cooling, core spray, the residual heat removal system, the service water and uninterrupted power supply. The initial request (Ref. 1) was based on an unquantified expectation that the unavailability for all these systems will improve with the elimination of the prescribed daily tests. The staff requested a quantification of the unavailability to justify the claimed improvement. The licensee submitted the additional information in Reference 2 which is a joint report with Pickard, Lowe and Garrick, Inc., and is based on probabilistic risk assessment methods with generic and plant-specific data. In Reference 6 the licensee added to the proposed change a requirement that within 24 hours following an ECCS failure the redundant train of ECCS be tested and within 24 hours following an SLC component failure the redundant SLC component be tested. The staff reviewed the proposal and the methodology and the data bases for the quantification of the unavailability. Our review and evaluation follows.

2.0 EVALUATION

2.1 Introduction

The current Vermont Yankee technical specifications require alternate testing of ECCS and SLC system or subsystems when other engineered safeguards systems or subsystems are out of service. These tests are required immediately when a system/subsystem is declared inoperable and daily thereafter. The following technical specifications and systems are involved:

Technical Specification	Inoperable System/Subsystem
4.4.B	Standby Liquid Control
4.5.A.2	Core Spray
4.5.A.3	Low Pressure Core Injection Pump
4.5.A.4	Low Pressure Core Injection System
4.5.B.2	Containment Cooling
4.5.C.2	Residual Heat Removal Service Water
4.5.C.3	Residual Heat Removal System
4.5.D.2	Service Water
4.5.D.3	Alternate Cooling
4.5.E.2	High Pressure Core Injection
4.5.F.2	Automatic Depressurization System
4.5.G.2	Reactor Core Isolation Cooling
4.5.H.1	Diesels
4.5.B.3.C	Standby Gas Treatment System
4.10.B.4	Uninterrupted Power Supply System

2.2 Types of Component and System Unavailabilities

In the context of this analysis, the term unavailability means the probability that a component or system is unable to accomplish its intended function. The causes of component unavailabilities are: failure, test override, repair, scheduled maintenance and human error. System unavailability can be caused by component unavailability and/or system alignment. Component failures can be time-related, demand-related or test-related. Correspondingly, systems have time, demand or test-related failures.

The licensee submitted quantification of the unavailabilities for the systems to perform their intended function upon demand. Two systems were chosen for detailed analysis: Core Spray System and the Diesel Generators. The other systems involved were reviewed in the context of the results obtained for the components of the Core Spray and the Diesel Generators.

The benefits and drawbacks of testing have been quantified; i.e., the decreased potential for an undetected failure and the increased unavailability and unavailability due to repair of demand-related and test-related failures. However, other alternate testing drawbacks such as:

- reduced reliability due to degradation from testing
- potential of plant transients initiated from surveillance testing
- potential for plant shutdown due to such transients
- diversion of maintenance personnel for testing, and
- potential increase in personnel radiation exposure from testing

have not been quantified; thus, the estimated benefit is conservative.

2.3 The Core Spray System

The system provides two trains of core spray for cooling during a loss-of-coolant accident. The main active components include the core spray pump and the following valves: pump suction, manual pump suction, test bypass, pump discharge bypass, inboard discharge, outboard discharge and manual isolation. The system alignments are: standby, injection, flow test and valve test.

Three data bases were used in this study:

- Generic data from a Pickard, Lowe and Garrick, LWR data base, Reference 3
- Vermont Yankee plant-specific data, and
- Human error data from NUREG/CR-1278 (Ref. 4)

The plant-specific demands and observed failures are small; thus, general data were "updated" with plant-specific data using Bayesian calculations (Ref. 3). The result is a probability of failure (unavailability) distribution that reflects the Vermont Yankee failure data base. In this manner, it was shown that for time and demand-related failures, daily testing produced unavailabilities at least a factor of 4 greater for daily testing versus monthly testing. Similar results were obtained for time and test-related failures.

In addition, the effect of the test interval change on the unavailability of both core spray trains was studied with and without common cause failures. The results again show unavailability improvement by a factor of about 4.

2.4 Diesel Generators

The same method was applied to estimate the diesel generator unavailability versus the test frequency. The generic data base was updated to reflect the observed Vermont Yankee failure rates.

The results for a single diesel, show a reduction of unavailability by a factor of about 3. More importantly, however, they show that the improvement is due to the reduction of test-induced failures. This is clear evidence against daily testing of diesel generators. Similar conclusions have been drawn from industry studies of fast starts of diesel generators which caused accelerated degradation (Ref. 5).

2.5 Application to Other Systems

As we have seen above, there are more systems involved in this application than the core spray and the diesel generators. The remaining systems can be classified into two functional categories: those not related to supply of water: uninterruptible power supply, the automatic depressurization and the standby gas treatment and all others which either pump or route cooling or borated water.

- Uninterruptible Power Supply (UPS)

The UPS provides ac power to the inverter when ac power is not available for the operation of the low pressure core injection valve. However, the battery banks are reliable and are not subject to the alternate ECCS requirements. The valve operation system is discussed below:

- Automatic Depressurization System (ADS)

When the high pressure core injection is declared inoperable, the ADS must be tested immediately and daily thereafter. Because the valves cannot be tested without causing a trip, only the ADS logic is subjected to the test. However, the actuation logic is very reliable. Under normal conditions, the ADS logic test is performed once each operating cycle. During the test, the valves must be deenergized, thus, increasing ADS unavailability, without benefit from the test.

° Standby Gas Treatment System (SGTS)

The SGTS handles gas with fans and dampers for which the generic failure rates are equivalent to those of the pumps and motor-operated valves of the core spray system (Ref. 3). Therefore, it is expected that it should behave in a similar manner with respect to the effect from test frequency (i.e., higher unavailability for daily versus monthly testing).

° Other Water Pumping and Routing Systems

In these types of systems, their unavailability is dominated by the failure of active components, i.e., pumps and valves. Therefore, these systems behave in a manner similar to the core spray we have examined above. Because the failure rates of their components are similar to those of the core spray it is reasonable to conclude that their unavailability decreases with decreasing testing frequency.

We have reviewed the information submitted by Vermont Yankee pertaining to the quantification of the unavailability of ECCS and SLC versus testing frequency. The method was based only on test and demand-related unavailabilities, thus, is conservative. The estimates were based on plant-specific modifications of generic data bases. Two systems (i.e., core spray and diesel generators) were explicitly analyzed. In addition, similarities were established for the other systems so that conclusions could be drawn regarding the expected behavior of system unavailability versus testing frequency. We find the methods and the data bases employed for these estimates to be acceptable.

The results of the licensee's study demonstrated that the unavailability of the SLC and ECCS systems decreases by a factor of at least 3 for a change of testing frequency from daily to monthly. We conclude that the elimination of the alternate testing requirements for Vermont Yankee will contribute to the increase of the SLC and ECCS system reliability; therefore, we find the proposed changes acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

Notice of Consideration by the staff of issuance of the proposed amendment was published in the Federal Register on January 26, 1988 (53 FR 2114) and requests for a hearing were received from the State of Vermont and the Commonwealth of Massachusetts. By a filing dated May 15, 1989, to the Atomic Safety and Licensing Board, the two intervenors and the licensee filed a joint motion to dismiss the proceeding. The Board granted the motion to dismiss in an Order dated May 23, 1989. An Environmental Assessment (EA) and Finding of No Significant Impact was published in the Federal Register on July 21, 1989 (54 FR 30619). Based upon the EA, the staff has determined not to prepare an environmental impact statement for the proposed license amendment, and has concluded that the proposed action will not have a significant adverse 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter from W. P. Murphy, Vermont Yankee Nuclear Power Corporation to USNRC, "Surveillance Testing of the ECCS and the SLC Equipment," (FVY 87-112), dated December 7, 1987.

2. Letter from R. W. Capstick, Vermont Yankee Corporation to USNRC, "VY Response to RAI - Surveillance Testing of ECCS and SLC Equipment," (FVY 88-58), dated July 15, 1988.

3. PLG-500, "Probabilistic Risk Assessment Data Base for Light Water Reactors," Pickard, Lowe and Garrick, Inc., to be published August 1988.

4. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," by A. D. Swain and H. H. Guttman.

5. NUREG-1024, "Technical Specifications - Enhancing the Safety Impact," prepared by the task group on Technical Specification, dated November 1983.

6. Letter from W. P. Murphy, Vermont Yankee Nuclear Power Corporation, to USNRC, "Surveillance Testing of ECCS and SLC Equipment, Supplement 2 to Proposed Change No. 85," (BVY 89-49), dated July 8, 1989.

Principal Contributor: L. Lois

Dated: July 21, 1989

UNITED STATES NUCLEAR REGULATORY COMMISSION

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 114 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 located in Windham County, Vermont. The amendment was effective as of the date of issuance.

The amendment revised the Technical Specifications to eliminate the present requirements to test the remaining train(s) of the ECCS and SLC systems immediately and daily thereafter when one train has a component out of service.

The application for amendment complies 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 is set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the Federal Register on January 26, 1988 (53 FR 2114). A request for a hearing was received from the State of Vermont and the Commonwealth of Massachusetts. Subsequently, the two intervenors and the applicant filed a joint motion to dismiss the proceeding.

The Atomic Safety and Licensing Board granted the motion to dismiss in an Order dated May 23, 1989.

The Commission has prepared an Environmental Assessment and Finding of No Significant Impact (54 FR 30619) related to the action and has concluded that an environmental impact statement is not warranted and that the issuance of this amendment will not have a significant adverse effect on the quality of the human environment.

For further details with respect to the action, see (1) the application for amendment dated December 7, 1987 and clarified by letters dated July 15, 1988 and June 8, 1989, (2) Amendment No. 114 to License No. DPR-28, and (3) the Commission's related Safety Evaluation and Environmental Assessment.

All of these items are available for public inspection at the Commission's Public Document Room, Gelman Building, Lower Level, 2120 L Street, N.W., Washington, D.C. and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects I/II.

Dated at Rockville, Maryland, this 21st day of July 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By

Morton B. Fairtile, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

PDI-3/LA	PDI-3/PM	OGC	PDI-3/DIR
MRushbrook	MBFairtile	C Bachmann	RHWessman
7/10/89	7/10/89	7/13/89	7/18/89