Docket No. 50-271

March 4, 1991

Mr. L. A. Tremblay
Senior Licensing Engineer
Vermont Yankee Nuclear Power Corporation
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Tremblay:

SUBJECT: ISSUANCE OF AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NO. DPR-28 - VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. 77221)

The Commission has issued the enclosed Amendment No. 128 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment is in response to your application dated July 20, 1990.

This amendment revises the surveillance testing requirements of certain engineered safeguards equipment in the Technical Specifications. An obsolete Technical Specification was deleted.

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

This completes action under TAC 77221.

Sincerely,

/S/

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

Enclosures:
1. Amendment No. 128 to
   License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous concurrence

LA:PDI-3	PM:PDI-3	OGC*	PDI-3/D(A)
MRushbrook	MBFairtile	SUttal	SFShankman
2/28/91	3/4/91	2/8/91	3/4/91

OFFICIAL RECORD COPY
[AMEND 77221]

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PDR ADOCK 05000271
P PDR

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

Robert M. Lazo, Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. John L. Lovering, Acting Director
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ATTN: James Muckerheide

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Administrative Judge
Atomic Safety and Licensing Board
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Washington, D.C. 20555

Jerry R. Kline
Administrative Judge
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 4, 1991

Docket No. 50-271

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Senior Licensing Engineer
Vermont Yankee Nuclear Power Corporation
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Tremblay:

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

This completes action under TAC 77221.

Sincerely,

for

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

Enclosures:
1. Amendment No. 128 to License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

AMENDMENT NO. 128 TO DPR-28 VERMONT YANKEE NUCLEAR POWER STATION DATED March 4, 1991

DISTRIBUTION:
Docket File 50-271
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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. 128
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 July 20, 1990, 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:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 128, 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

Susan F. Shankman, Acting 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: March 4, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 128

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
79	79
83	83
84	84
85	85
86	86
87	87
88	88
89	89
90	90
91	91
92R	92R
93	93
95	95
99	99
100	100
103	103
104	104
108	108
122	122
122a	122a (blank page)
123	123
127	127
128	128
129	129
131	131
132	132
133	133
143	143
144	144
146	146
173	173
180	180

3.4 LIMITING CONDITIONS FOR OPERATION

3.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the operating status of the Reactor Standby Liquid Control System.

Objective

To assure the availability of an independent reactivity control mechanism.

Specification

A. Normal Operation

Except as specified in 3.4.B below, the Standby Liquid Control System shall be operable during periods when fuel is in the reactor unless:

1. The reactor is in cold shutdown

and

2. Control rods are fully inserted and Specification 3.3.A is met.

4.4 SURVEILLANCE REQUIREMENTS

4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Applicability

Applies to the periodic testing requirement for the Reactor Standby Liquid Control System.

Objective

To verify the operability of the Standby Liquid Control System.

Specification

A. Normal Operation

The Standby Liquid Control System shall be verified operable by:

1. Testing pumps and valves in accordance with Specification 4.6.E. A minimum flow rate of 35 gpm at 1275 psig shall be verified for each pump by recirculating demineralized water to the test tank.

2. Verifying the continuity of the explosive charges at least monthly.

In addition, at least once during each operating cycle, the Standby Liquid Control System shall be verified operable by:

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. Pump operability testing in accordance with Specification 4.6.E 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.

3.4 & 4.4 REACTOR STANDBY LIQUID CONTROL SYSTEM

Once the solution has been made up, boron concentration will not vary unless more boron or more water is added. Level indication and alarm indicate whether the solution volume has changed which might indicate a possible solution concentration change. Considering these factors, the test interval has been established.

Sodium pentaborate concentration is determined within 24 hours following the addition of water or boron, or if the solution temperature drops below specified limits. The 24-hour limit allows for 8 hours of mixing, subsequent testing, and notification of shift.

3.5 LIMITING CONDITION FOR OPERATION

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applies to the operational status of the Emergency Cooling Subsystems.

Objective:

To assure adequate cooling capability for heat removal in the event of a loss-of-coolant accident or isolation from the normal reactor heat sink.

Specification:

A. Core Spray and Low Pressure Coolant Injection

1. Except as specified in Specifications 3.5.A.2 through 3.5.A.4 below and 3.5.H.3 and 3.5.H.4, both Core Spray and the LPCI Subsystems shall be operable whenever irradiated fuel is in the reactor vessel and prior to a reactor startup from the cold shutdown condition.

4.5 SURVEILLANCE REQUIREMENT

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability:

Applied to periodic testing of the emergency cooling subsystems.

Objective:

To verify the operability of the core containment cooling subsystems.

Specification:

A. Core Spray and Low Pressure Cooling Injection

Surveillance of the Core Spray and LPCI Subsystems shall be performed as follows:

1. General Testing

Item	Frequency
a. Simulated Automatic Actuation Test	Each re-fueling outage
b. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.	
c. Flow Rate Test-Core Spray pumps shall deliver at least 3000 gpm (torus to torus) against a system head of 120 psig. Each LPCI pump shall deliver 7450 ± 150 gpm (vessel to vessel).	Each re-fueling outage

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.

4.5 SURVEILLANCE REQUIREMENT

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.

3.5 LIMITING CONDITION FOR OPERATION

4. a. From and after the date that a LPCI Subsystem is made or found to be inoperable due to failure of the associated UPS, reactor operation is permissible only during the succeeding thirty days, for the 1989/90 operating cycle, unless it is sooner made operable, provided that during that time the associated motor control center (89A or 89B) is powered from its respective maintenance tie, all active components of the other LPCI and the Containment Cooling Subsystem, the Core Spray Subsystems, and the emergency diesel generators shall be operable, the requirements of Specification 3.10.A.4 are met, and the 4160 volt tie line to the Vernon Hydro is the operable delayed access power source.

   b. From and after the date that a LPCI Subsystem is made or found to be inoperable for any reason, other than failure of the UPS during the 1989/90 operating cycle, or Specification 3.5.A.4.a is not met, reactor operation is permissible only during the succeeding seven days unless it is sooner made operable, provided that during that time 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 within 24 hours.

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. Recirculation pump discharge valves shall be tested to verify full open to full closed in $27 \leq t \leq 33$ seconds and bypass valves shall be tested for operability in accordance with Specification 4.6.E.

3.5 LIMITING CONDITION FOR OPERATION

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 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

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.

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:

   Operability testing of pumps and valves shall be in accordance with Specification 4.6.E.

   Each RHR service water 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. Station Service Water and Alternate Cooling Tower Systems

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

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. Station Service Water and Alternate Cooling Tower Systems

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

1. Operability testing of pumps and valves shall be in accordance with Specification 4.6.E. Each pump shall deliver at least 2700 gpm against a TDH of 250 feet.

3.5 LIMITING CONDITION FOR OPERATION

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

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 Cooling 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 System shall be performed as follows:

1. Testing:

Item	Frequency
Simulated Automatic Actuation Test	Each refueling outage

   Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The HPCI System shall deliver at least 4250 gpm at normal reactor operating pressure when recirculating to the Condensate Storage Tank.

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. Operability testing of the relief valves shall be in accordance with Specification 4.6.E.

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
Simulated automatic actuation test (testing valve operability)	Each refueling outage

Operability testing of the pump and valves shall be in accordance with Specification 4.6.E. The RCIC System shall deliver at least 400 gpm at normal operating pressure when recirculating to the Condensate Storage Tank.

3.5 LIMITING CONDITION FOR OPERATION

4. When irradiated fuel is in the reactor vessel and the reactor is in the refueling condition, both LPCI subsystems, or both Core Spray systems, or one diesel generator may be inoperable provided that a source of water of greater than 300,000 gal. is available to the operable core cooling subsystem.

I. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystem, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

4.5 SURVEILLANCE REQUIREMENT

I. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to to assure that the discharge piping of the core spray subsystems, LPCI subsystem, HPCI and RCIC are filled:

1. Every month and prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point and water flow observed.

2. Following any period where the LPCI subsystem or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI or RCIC system is lined up to take suction from the torus, the discharge piping of the HPCI and RCIC shall be vented from the high point of the system and water flow observed on a monthly basis.

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 referenced in Specification 4.6.E. 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.

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.

3.5 (cont'd)

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 referenced in Specification 4.6.E. 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 COOLANT SYSTEMS

A. Core Spray and LPCI

During normal plant operation, manual tests of operable pumps and valves shall be conducted in accordance with Specification 4.6.E to demonstrate operability.

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

Periodic testing as described in Specification 4.6.E 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 requirements 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 COOLING 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 described in Specification 4.6.E 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.

G. Reactor Core Isolation Cooling System

Frequency of testing of the RCIC System is the same as the HPCI, per Specification 4.6.E, 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 described in Specification 4.6.E. 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 as discussed in Section I 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.

3.6 LIMITING CONDITIONS FOR OPERATION

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

2. Both the sump and air sampling systems shall be operable during power operation. From and after the date that one of these systems is made or found inoperable for any reason, reactor operation is permissible only during succeeding seven days.

3. If these conditions cannot be met, initiate an orderly shutdown and the reactor shall be in the cold shutdown condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 120 psig and temperature greater than 350°F, both safety valves shall be operable. The relief valves shall be operable, except that if one relief valve is inoperable, reactor power shall be immediately reduced to and maintained at or below 95% of rated power.

2. If Specification 3.6.D.1 is not met, initiate an orderly shutdown and the reactor coolant pressure shall be below 120 psig and 350°F within 24 hours.

4.6 SURVEILLANCE REQUIREMENTS

C. Coolant Leakage

Reactor coolant system leakage shall be checked and logged at least once per day.

D. Safety and Relief Valves

1. Operability testing of Safety and Relief Valves shall be in accordance with Specification 4.6.E. The lift point of the safety and relief valves shall be set as specified in Specification 2.2.B.

3.6 & 4.6 (Continued)

greater than the limit specified for unidentified leakage; the probability is small that imperfections or cracks associated with such leakage would grow rapidly. Leakage less than the limit specified can be detected within a few hours utilizing the available leakage detection systems. If the limit is exceeded and the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

The removal capacity from the drywell floor drain sump and the equipment drain sump is 50 gpm each. Removal of 50 gpm from either of these sumps can be accomplished with considerable margin.

D. Safety and Relief Valves

Parametric evaluations have shown that only three of the four relief valves are required to provide a pressure margin greater than the recommended 25 psi below the safety valve actuation settings as well as a MCPR > 1.06 for the limiting overpressure transient below 98% power. Consequently, 95% power has been selected as a limiting power level for three valve operation. For the purposes of this limiting condition a relief valve that is unable to actuate within tolerance of its set pressure is considered to be as inoperable as a mechanically malfunctioning valve.

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ± 1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded.

E. Structural Integrity and Operability Testing

A pre-service inspection of the components listed in Table 4.2-4 of the FSAR was conducted after site erection to assure freedom from defects greater than code allowance; in addition, this serves as a reference base for further inspections. Prior to operation, the reactor primary system was free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout plant life. The inservice inspection and testing programs are performed in accordance with 10CFR50, Section 50.55a(g) except where specific relief has been granted by the NRC. These inspection and testing programs provide further assurance that gross defects are not occurring and ensure that safety-related components remain operable.

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3.6 & 4 (CONT'D)

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is sensitive, fast, and reliable. Magnetic particle and liquid penetrant inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing and radiography shall be used where defects can occur on concealed surfaces.

The in-service inspection and testing programs presented at this time are based on a thorough evaluation of present technology and state-of-the-art inspection and testing techniques.

F. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ±5%, the flow rates in both recirculation loops will be verified by main Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured value of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the measured value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measure. This decrease, together with the loop flow increase, would result in a leak of correlation between measured and derived core flow rate.

3.7 LIMITING CONDITIONS FOR OPERATION

3. Whenever primary containment is required, the total primary containment leakage rate shall not exceed 0.8 weight percent per day (L_a) at a pressure of 44 psig (P_a).

4. Whenever primary containment is required, the leakage from any one isolation valve shall not exceed 5 percent of the maximum allowable leak rate (L_a) at peak accident pressure (P_a) and the leakage from any one main steam line isolation valve shall not exceed 15.5 scf/hr at 44 psig (P_a).

5. <u>Pressure Suppression Chamber-Reactor Building Vacuum Breakers</u>

   a. Two of two pressure suppression chamber-reactor building vacuum breaker systems shall be operable at all times when the primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air-operated vacuum breakers shall be $\leq$0.5 psid. The self actuating vacuum breakers shall open fully when subjected to a force equivalent to or less than 0.5 psid acting on the valve disk.

   b. With one Reactor Building - Suppression chamber vacuum breaker inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within seven (7) days or then be in cold shutdown within the following 24 hours.

   c. With one Reactor Building - Suppression chamber vacuum breaker failed open - power operation may continue provided the other vacuum breaker in that line is verified to be closed and conditions required by 3.7.D.2 are met.

4.7 SURVEILLANCE REQUIREMENTS

3. Prior to violating the integrity of a system outside the primary containment, which is connected to any valve listed in Table 4.7.2b, the isolation valves bounding the opening shall have Type C tests performed. If the opening cannot be isolated from the containment by two isolation valves which meet the acceptance criteria of Appendix J (10CFR Part 50), a blank flange shall be installed on the opening.

4. The leakage from any one isolation valve shall not exceed 5% of Ltm. The leakage from any one main steam line isolation valve shall not exceed 11.5 scf/hr at 24 psig (Pt). Repair and retest shall be conducted to insure compliance.

5. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u>

   a. The pressure suppression chamber-Reactor Building vacuum breaker instrumentation including setpoint shall be checked for proper operation every three months.

   b. Operability testing of the vacuum breakers shall be in accordance with Specification 4.6.E. Each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.5.a and each vacuum breaker shall be inspected and verified to meet design requirements.

3.7 LIMITING CONDITIONS FOR OPERATION

6. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specifications 3.7.A.6.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:

      (1) The valve is demonstrated to open fully with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

      (2) The valve can be closed by gravity, when released after being opened by remote or manual means, to within not greater than the equivalent of 0.05 inch at all points along the seal surface of the disk.

      (3) The position alarm system will annunciate in the control room if the valve opening exceeds the equivalent of 0.05 inch at all points along the seal surface of the disk.

   b. Up to two (2) of the ten (10) suppression chamber - drywell vacuum breakers may be determined to be inoperable provided that they are secured, or known to be, in the closed position.

4.7 SURVEILLANCE REQUIREMENTS

6. Pressure Suppression Chamber - Drywell Vacuum Breakers

   a. Periodic Operability Tests

      Operability testing of the vacuum breakers shall be in accordance with Specification 4.6.E and following any release of energy to the suppression chamber. Operability of the corresponding position switches and position indicators and alarms shall be verified monthly and following any maintenance.

   b. Refueling Outage Tests

      (1) All suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

      (2) At least two (2) of the suppression chamber - drywell vacuum breakers shall be inspected. If deficiencies are found such that Specification 3.7.A.6 could not be met, all vacuum breakers shall be inspected and deficiencies corrected.

3.7 LIMITING CONDITIONS FOR OPERATION

c. Reactor operation may continue for fifteen (15) days provided that at least one position alarm circuit for each vacuum breaker is operable and each suppression chamber - drywell vacuum breaker is physically verified to be closed immediately and daily thereafter.

7. Oxygen Concentration

a. The primary containment atmosphere shall be reduced to less than 4 percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 90 psig, except as specified in Specification 3.7.A.7.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

8. If Specification 3.7.A.1 through 3.7.A.7 cannot be met, an orderly shutdown shall be initiated immediately and the reactor shall be in a cold shutdown condition within 24 hours.

4.7 SURVEILLANCE REQUIREMENT

(3) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of the leakage rate through a 1-inch orifice.

7. Oxygen Concentration

The primary containment oxygen concentration shall be measured and recorded on a weekly basis.

3.7 LIMITING CONDITIONS FOR OPERATION

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. Operability testing of valves shall be in accordance with Specification 4.6.E.

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

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. The fuel cask or irradiated fuel is not being moved in the Reactor Building.

2. Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.

D. <u>Primary Containment Isolation Valves</u>

1. During reactor power operating conditions all isolation valves listed in Table 4.7.2 and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

4.7 SURVEILLANCE REQUIREMENTS

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain a 0.15 inch of water vacuum under calm wind ($2<\bar{u}<5$ mph) conditions with a filter train flow rate of not more than 1,500 cfm, shall be demonstrated at least quarterly and at each refueling outage prior to refueling.

2. The core spray and LPCI lower compartment openings shall be checked closed daily.

D. <u>Primary Containment Isolation Valves</u>

1. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E:

a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and the closure times specified in Table 4.7.2.

3.7 LIMITING CONDITIONS FOR OPERATION

2. In the event any isolation valve specified in Table 4.7.2 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS

b. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

c. At least twice per week, the main steam line isolation valves shall be exercised by partial closure and subsequent reopening.

2. Whenever an isolation valve listed in 4.7.2 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

4.7.A (Continued)

The leak rate testing program is based on AEC guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression Chamber-Reactor Building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. Operability testing is performed in conjunction with Specification 4.6.E. Inspections and calibrations are performed during the refueling outages; this frequency being based on equipment quality, experience, and engineering judgment.

The ten (10) drywell-suppression vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified. Also it is inspected to assure that it closes freely and operates properly.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.12 ft^2. This is equivalent to one vacuum breaker open by three-eighths of an inch (3/8") as measured at all points around the circumference of the disk or three-fourths of an inch (3/4") as measured at the bottom of the disk when the top of the disk is on the seat. Since these valves open in a manner that is purely neither mode, a conservative allowance of one-half inch (1/2") has been selected as the maximum permissible valve opening. Assuming that permissible valve opening could be evenly divided among all ten vacuum breakers at once, valve open position assumed to indication for an individual valve must be activated less than fifty-thousandths of an inch (0.050") at all points along the seal surface of the disk. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a nonseated valve.

At the end of each refueling cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure and held constant. The 2 psig set point will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure guage. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed. If the drywell pressure can be increased by 1 psi over the suppression chamber the rate of change of the suppression chamber pressure must not exceed a rate equivalent to the rate of leakage from the

4.7.A (Continued)

drywell through a 1-inch orifice. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised in accordance with Specification 4.6.E and immediately following termination of discharge of steam into the suppression chamber. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights are designed to function as follows:

Full Closed (Closed to ≤0.050" open)	2 White - On
Open (>0.050" open to full open)	2 White - Off

During each refueling outage, two drywell-suppression chamber vacuum breakers will be inspected to assure sealing surfaces and components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in one-eighth of the design lifetime is extremely conservative.

Experience has shown that a weekly measurement of the oxygen concentration in the primary containment assures adequate surveillance of the primary containment atmosphere.

B. and C. Standby Gas Treatment System and Secondary Containment System

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 0.15 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leakage tightness of the reactor building, and performance of the standby gas treatment system. Functionally testing of initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

4.7.D. Primary Containment Isolation Valves

Those large pipes comprising a portion of the reactor coolant system whose failure could result in uncovering the reactor core are supplied with automatic isolation valves (except those lines needed for emergency core cooling system operation or containment cooling). The closure times specified herein and per Specification 4.6.E are adequate to prevent loss of more cooling from the circumferential rupture of any of these lines outside the containment than from a steam line rupture. Therefore, the isolation valve closure times are sufficient to prevent uncovering the core.

Purge and vent valve testing performed by Allis-Chalmers has demonstrated that all butterfly purge and vent valves installed at Vermont Yankee can close from full open conditions at design basis containment pressure. However, as an additional conservative measure, limit stops have been added to valves 16-19-7/7A, limiting the opening of these valves to 50^{o} open while operating, as requested by NRC in their letter of May 22, 1984. (NVY 84-108)

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate the fuel rod cladding perforations would be avoided for the main steam valve closure times, including instrument delay, as long as 10.5 seconds. The test closure time limit of five seconds for these main steam isolation valves provides sufficient margin to assure that cladding perforations are avoided and 10CFR100 limits are not exceeded. Redundant valves in each line ensure that isolation will be effected applying the single failure criteria.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The containment is penetrated by a large number of small diameter instrument lines. The flow check valves in these lines are tested for operability in accordance with Specification 4.6.E.

3.10 LIMITING CONDITIONS FOR OPERATION

3.10 Auxiliary Electrical Power Systems

Applicability

Applies to the auxiliary electrical power systems.

Objective

To assure an adequate supply of electrical power for operation of those systems required for reactor safety.

Specification

A. Normal Operation

The reactor shall not be made critical unless all of the following conditions are satisfied.

1. Diesel Generators

Both emergency diesel generators shall be operable and capable of starting and reaching rated voltage and frequency in not more than 13 seconds.

4.10 SURVEILLANCE REQUIREMENTS

4.10 Auxiliary Electrical Power Systems

Applicability

Applies to the periodic testing requirements of the auxiliary electrical power systems.

Objective

To verify the operability of the auxiliary electrical power systems.

Specification

A. Normal Operation

1. Diesel Generators

a. Each diesel generator shall be started and loaded once a month to demonstrate operational readiness. The test shall continue until the diesel engine and the generator are at equilibrium temperature at expected maximum emergency loading not to exceed the continuous rating. During this test, the diesel starting time to reach rated voltage and frequency shall be logged and the air compressor shall be checked for operation and its ability to recharge air receivers. The diesel fuel oil transfer pumps shall be tested in accordance with Specification 4.6.E.

4.10.A (Continued)

Both diesel generators have air compressors and air receivers tanks for starting. It is expected that the air compressors will run only infrequently. During the monthly check of the units, each receiver will be drawn down below the point at which the compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

Following the tests of the units and at least weekly, the fuel volume remaining will be checked. At the end of the monthly load test of the diesel generators, the fuel oil transfer pump will be operated to refill the day tank. The day tank level indicator and alarm switches will be checked at this time. Fuel oil transfer pump operability testing is in accordance with Specification 4.6.E.

The test of the diesels and Uninterruptible Power Systems during each refueling interval will be more comprehensive in that it will functionally test the system; i.e., it will check starting and closure of breakers and sequencing of loads. The units will be started by simulation of a loss of coolant accident. In addition, a loss of normal power condition will be imposed to simulate a loss of off-site power. The timing sequence will be checked to assure proper loading in the time required. Periodic tests between refueling intervals check the capability of the diesels to start in the required time and to deliver the expected emergency load requirements. Periodic testing of the various components plus a functional test at a refueling interval are sufficient to maintain adequate reliability.

B. Although the Main Station, ECCS, AS-2, and UPS batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure.

The performance discharge test provides adequate indication and assurance that the batteries have the specified ampere hour capacity. The rate of discharge during this test shall be in accordance with the manufacturer's discharge characteristic curves for the associated batteries. The results of these tests will be logged and compared with the manufacturer's recommendations of acceptability.

The service discharge test provides a test of the batteries ability to satisfy the design requirements (battery duty cycle) of the associated dc system. This test will be performed using simulated or actual loads at the rates and for the durations specified in the design load profile.

C. Logging the diesel fuel supply weekly and after each operation assures that the minimum fuel supply requirements will be maintained. During the monthly test for quality of the diesel fuel oil, a viscosity test and water and sediment test will be performed as described in ASTM D975-68. The quality of the diesel fuel oil will be acceptable if the results of the tests are within the limiting requirements for diesel fuel oils shown on Table 1 of ASTM D975-68.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 128 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

INTRODUCTION

By letter dated July 20, 1990, the Vermont Yankee Nuclear Power Corporation (the licensee) requested an amendment to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The proposed amendment would revise the surveillance testing requirements of certain engineered safeguards equipment in the Technical Specifications (TS) and deletes an obsolete requirement from the T.S.

DISCUSSION

Vermont Yankee has evaluated its current inservice testing program against the NRC Generic Letter 89-04 and as a consequence is proposing changes to the Technical Specifications concerning certain surveillances. The proposed changes will ensure that all inservice testing of pumps and valves will be conducted under American Society of Mechanical Engineers (ASME) Code Section XI. In addition, Vermont Yankee is requesting the deletion of a portion of Technical Specification Sections 3.5.A.3 and 4.5.A.3 that concern extending the limiting condition of operation to facilitate the replacement of impeller wear rings on the RHR pumps. These replacements were made during the 1986-1987 operating cycle thus this portion of the TS is now obsolete.

The systems with changes to the surveillance requirements are the following:

a) Reactor Standby Liquid Control System

b) Core Spray and Low Pressure Cooling Injection System

c) Residual Heat Removal Service Water System

d) Station Service Water and Alternate Cooling Water System

e) High Pressure Coolant Injection System

f) Automatic Depressurization System

g) Reactor Core Isolation Cooling System

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h) Pressure Suppression Chamber - Reactor Building Vacuum Breakers

i) Pressure Suppression Chamber - Drywell Vacuum Breakers

j) Primary Containment Isolation Valves

k) Diesel Fuel Oil Transfer Pumps

The proposed change removes individual surveillance frequencies from the noted systems and substitutes a reference to Technical Specification Section 4.6.E, Structural Integrity and Operability Testing. Section 4.6.E states that inservice inspection and testing programs are to be performed in accordance with Section XI of the ASME code in accordance with 10 CFR 50.55a(g).

EVALUATION

The current testing frequency of the applicable pumps and valves listed above, except for a few that are tested at intervals of three months or greater, is once per month. ASME Code Section XI permits this testing to be performed once every three months unless testing is not practical during plant operation. In that case the relevant components must be identified by the licensee and tested only during cold shutdown periods. ASME Code Section XI is included by reference in 10 CFR 50.55a.

Based on the above we conclude that the surveillance interval changes are acceptable. The changes to TS 3.5.A.3 and 4.5.A.3 are also acceptable as they merely delete an obsolete requirement.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 36357) on September 5, 1990 and consulted with the State of Vermont. No public comments were received and the State of Vermont did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: March 4, 1991