OCT 8 1975

Docket No. 50-271

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Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 16 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment includes Change No. 27 to the Technical Specifications and is in response to Vermont Yankee's request dated March 31, 1975, which was submitted in reply to our letter dated February 14, 1975.

This amendment incorporates: (1) water temperature limits during any testing which adds heat to the suppression pool, (2) suppression pool water temperature limits requiring manual scram of the reactor, (3) suppression pool water temperature limits requiring reactor pressure vessel depressurization, (4) surveillance requirements to monitor water temperatures during operations which add heat to the suppression pool and (5) external visual examinations of the suppression chambers following operations in which the pool temperatures exceed 160°F.

A copy of the Federal Register Notice relating to this action is also enclosed.

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Enclosures:
1. Amendment No. 16
w/Change No. 27
2. Federal Register Notice

cc: See next page

cc-4

OFFICE ➤	ORB#4	ORB#4	OELD	ORB#4		
SURNAME ➤	RIngram/dg	DiBenedetto	Gitner	RWReid		
DATE ➤	9/30/75	9/30/75	10/6/75	10/8/75		

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

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cc: See next page

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Putney, Vermont 05346

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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. 16
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated March 31, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Facility License No. DPR-28 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 27."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

Attachment:
Change No. 27 to the
Technical Specifications

Date of Issuance: OCT 3 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 16

CHANGE NO. 27 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Delete pages 126, 139 and 141 from the Appendix A Technical Specifications and insert the attached replacement pages 126, 126a, 139, 139a, 141 and 141a. The changed areas on the revised pages are shown by marginal lines.

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Whenever primary containment is required, the volume and temperature of the water in the suppression chamber shall be maintained within the following limits:

   a. Maximum Water Temperature during normal operation - 90°F.

   b. Maximum Water Temperature during any test operation which adds heat to the suppression pool - 100°F and shall not be above 90°F for more than 24 hours. 27

   c. If Torus Water Temperature exceeds 110°F, initiate an immediate scram of the reactor. Power operation shall not be resumed until the pool temperature is reduced below 90°F. 27

   d. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the torus water temperature exceeds 120°F. 27

4.7 SURVEILLANCE REQUIREMENTS

4.7 STATION CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment system integrity.

Objective:

To verify the integrity of the primary and secondary containments.

Specification:

A. Primary Containment

1. The suppression chamber water level and temperature shall be checked once per day. A visual inspection of the suppression chamber interior including water line regions and the interior painted surfaces above the water line shall be made at each refueling outage. Whenever there is indication of relief valve operation which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation. 27

3.7 LIMITING CONDITIONS FOR OPERATION

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 78,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

4.7 SURVEILLANCE REQUIREMENTS

4.7 STATION CONTAINMENT SYSTEMS

2. The primary containment integrity shall be demonstrated as required by Appendix J to 10 CFR Part 50. The primary containment shall meet the containment acceptance requirements set forth in that appendix.

   a. Penetrations and seals listed in Table 4.7.1 shall be leak tested at 44 psig (Pa).

   b. Type C tests shall be performed on the isolation valves listed in Table 4.7.2.a.

3.7.A (cont'd)

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 90°F will assure that the 170°F limit is not approached.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures 27
define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Double isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and suppression chamber and reactor building so that the structural integrity of the containment is maintained.

The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure is 2 psig.

The capacity of the ten (10) drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows eight (8) operable

3.7.A (cont'd)

valves, therefore, with two (2) valves secured, containment integrity is not impaired.

Each drywell-suppression chamber vacuum breaker is fitted with a redundant pair of limit switches to provide fail safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.050" at all points along the seal surface of the disk. These switches are capable of transmitting the disk closed to open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundance of components and circuits and, most importantly, the accessability of the valve lever arm and position reference external to the valve. The fail safe feature of the alarm circuits assures operator attention if a line fault occurs.

---

(1) Robbins, C. H., Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment", GEAP-3596, November 17, 1960.

(2) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

(3) CodeAllowable peak accident pressure is 62 psig.

3.7.D Primary Containment Isolation Valves

Double isolation valves are provided on lines that penetrate the primary containment and communicate directly with the reactor vessel and on lines that penetrate the primary containment and communicate with the primary containment free space. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

4.7 STATION CONTAINMENT SYSTEMS

A. Primary Containment System

The water in the suppression chamber is used only for cooling in the event of an accident, i.e., it is not used for normal operation; therefore, a weekly check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature
normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature
trends. By requiring the suppression pool temperature to be continually monitored and frequently logged
during periods of significant heat addition, the temperature trends will be closely followed so that
appropriate action can be taken. The requirement for an external visual examination following any 27
event where potentially high loadings could occur provides assurance that no significant damage was
encountered. Particular attention should be focused on structural discontinuities in the vicinity of
the relief valve discharge since these are expected to be the points of highest stress. Visual
inspection of the suppression chamber including water line regions each refueling outage is adequate
to detect any changes in the suppression chamber structures.

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 44 psig which would rapidly reduce to 27 psig within 10 seconds following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig within 10 seconds, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay.[1]

(1) Section 5.2 of the FSAR.

4.7.A (cont'd)

The design pressure of the drywell and absorption chamber is 56 psig.[2] The design leak rate is 0.5%/day at a pressure of 62 psig. As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

(2) 62 psig is the maximum allowable peak accident pressure for this design (56 psig) pressure.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 16 to Facility Operating License No. DPR-28 issued to Vermont Yankee Nuclear Power Corporation (the licensee) which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility) located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment incorporates additional suppression pool water temperature limits: (1) during any testing which adds heat to the pool, (2) at which reactor scram is to be initiated and (3) requiring reactor pressure vessel depressurization. It also adds surveillance requirements for visual examination of the suppression chamber during each refueling and following operations in which the pool temperatures exceed 160°F and adds monitoring requirements of water temperatures during operations which add heat to the pool.

The application for the 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 are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on July 23, 1975 (40 FR 30886). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

For further details with respect to this action, see (1) the application for amendment dated March 31, 1975, (2) Amendment No. 16 to License No. DPR-28, with Change No. 27 and (3) the Commission's related Safety Evaluation issued July 15, 1975. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H 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, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland
this 8th day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing