June 1, 1994

Docket No. 50-271

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Mr. Donald A. Reid, Vice President
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Vermont Yankee Nuclear Power Corporation
Ferry Road
Brattleboro, Vermont 05301

Dear Mr. Reid:

SUBJECT: ISSUANCE OF AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-28, VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. M 87041)

The Commission has issued the enclosed Amendment No. 139 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment is in response to your application dated July 14, 1993.

This amendment revises Sections 3.6 and 4.6 of the Technical Specifications to incorporate reactor coolant system leakage detection requirements to address Generic Letter 88-01 "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

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

Sincerely,

Original signed by:

Daniel H. Dorman, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 139 to License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

OTSB CG #94-103
CGrimes 5/26/94 Tolerable inconsistencies

OFFICE	PDI-3:LA	PDI-3:PM	EMCB	OGC	PDI-3:D
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DATE	4/18/94	4/18/94	5/5/94	5/13/94	6/1/94

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

June 1, 1994

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Vermont Yankee Nuclear Power Corporation
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Daniel H. Dorman

Daniel H. Dorman, Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
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   License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

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VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 139
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 14, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 139 , 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 and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, 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: June 1, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 139

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
108	108
108a	108a
122	122
123	123

VYNPS

3.6 LIMITING CONDITIONS FOR OPERATION

C. Coolant Leakage

1a. 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 leakage into the primary containment shall not exceed 25 gpm.

b. While in the run mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm within any 24 hour period.

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

1. Reactor coolant system leakage, for the purpose of satisfying Specification 3.6.C.1, shall be checked and logged once per shift, not to exceed 12 hours.

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.

VYNPS

3.6 LIMITING CONDITIONS FOR OPERATION

E. Structural Integrity and Operability Testing

The structural integrity and the operability of the safety-related systems and components shall be maintained at the level required by the original acceptance standards throughout the life of the plant.

4.6 SURVEILLANCE REQUIREMENTS

E. Structural Integrity and Operability Testing

1. Inservice inspection of safety-related components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter.

2. Operability testing of safety-related pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

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 2 gpm increase limit in any 24 hour period for unidentified leaks was established as an additional requirement to the 5 gpm limit by Generic Letter 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping."

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 purpose 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 will be 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 will be 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.

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.

Generic Letter 88-01 established the NRC position for in-service inspection of BWR austenitic stainless steel piping susceptible to Intergranular Stress Corrosion Cracking (IGSCC).

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. <u>Jet Pumps</u>

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

On January 25, 1988, the NRC issued Generic Letter (GL) 88-01, "NRC Position on IGSCC (Intergranular Stress Corrosion Cracking) in BWR Austenitic Stainless Steel Piping." Vermont Yankee Nuclear Power Corporation (VY, or the licensee) responded to the GL for the Vermont Yankee Nuclear Power Station (VYNPS) in a letter dated July 27, 1988. The NRC staff provided its Safety Evaluation (SE) for VYNPS in a letter dated February 14, 1990, in which the staff found the licensee response acceptable with the exceptions that VY incorporated leakage detection requirements in administrative procedures rather than in the plant Technical Specifications (TSs) as requested in the GL, and that the licensee's leakage detection procedure provided for averaging of detected leakage over a 24-hour period. By letter dated March 8, 1990, the licensee responded to the staff's SE stating its belief "that the combination of existing TS and administrative controls fully comply with the intent of the Staff's position on coolant leakage."

On February 4, 1992, the NRC issued GL 88-01, Supplement 1, in which the staff "determined that incorporation of the leakage detection requirements in an administrative document is not acceptable." By letter dated May 22, 1992, the NRC staff responded directly to VY's letter of March 8, 1990, reaffirming the unacceptability of the licensee's response regarding leakage detection requirements and requesting that VY submit a proposed TS change consistent with the GL. By letter dated September 21, 1992, VY claimed that the staff's request constituted a backfit as defined in 10 CFR 50.109 in that the cost could not be justified by any comparable safety improvement. Following discussions with the staff, the licensee withdrew the backfit claim in a letter dated October 27, 1992, and submitted a proposed TS change in a letter dated July 14, 1993. The staff confirmed the licensee's withdrawal of the backfit claim and reaffirmed its position in a letter to the licensee dated January 21, 1993.

The change proposed by VY in its letter dated July 14, 1993, would: (1) limit the increase in reactor coolant leakage into the primary containment from unidentified sources to not more than 2 gpm within any 24-hour period; (2) include a reference to GL 88-01 as the basis for the 2 gpm limit on increases in unidentified leakage; and (3) require compliance with GL 88-01 when performing the inservice inspection program for the piping identified in the GL.

2.0 EVALUATION

The licensee proposes to modify the TS as follows:

> Existing TS 3.6.C.1 is renumbered TS 3.6.C.1a and new TS 3.6.C.1b is added as follows: "While in the run mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm within any 24 hour period."
>
> TS 4.6.C.1 is revised to require checking and logging of reactor coolant system leakage "once per shift, not to exceed twelve hours" in lieu of "at least once per day."
>
> The following statement is appended to TS 4.6.E.1 regarding surveillance requirements for structural integrity: "Inservice inspection of piping, identified in NRC Generic Letter 88-01, shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in the Generic Letter."

Existing TS 3.6.C.3 provides the action statement requiring initiation of an orderly shutdown and placement of the reactor in the cold shutdown condition within 24 hours if "these conditions cannot be met." "These conditions" would include the new requirement of Specification 3.6.C.1b.

The proposed changes to TS 3.6.C.1 would establish a limit on the rate of increase of unidentified leakage during operations in the run mode and require initiation of a plant shutdown if such leakage increases by more than 2 gpm within any 24-hour period. The staff finds that this change is consistent with GL 88-01 and is therefore acceptable.

The proposed change to TS 4.6.C.1 would require that leakage measurements be taken once per shift not to exceed 12 hours. In GL 88-01, the staff requested that such measurements be taken every 4 hours. In Supplement 1, however, the staff found that "monitoring reactor coolant system (RCS) leakage every 4 hours creates an unnecessary administrative hardship for plant operators.

Thus, RCS leakage measurements should be taken at least once per shift, not to exceed 12 hours." The staff finds that the proposed change is consistent with the staff's position and is therefore acceptable.

The proposed change to TS 4.6.E.1 would add a statement regarding conduct of inservice inspection of piping susceptible to IGSCC as discussed in the GL. The staff finds that the statement is consistent with the guidance provided in GL 88-01, and is therefore acceptable.

The licensee has also proposed two changes to the TS Bases. Specifically, references to GL 88-01 are provided as the basis for the new requirements for leakage monitoring and inservice inspection in Specifications 3/4.6.C and 3.6.E. The staff has no objections to the proposed Bases changes.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Vermont State Official was notified of the proposed issuance of the amendment. The State Official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding of no significant hazards consideration, and there has been no public comment on such finding (59 FR 12370). Accordingly, the 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 the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: D. Dorman

Date: June 1, 1994