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

October 4, 1996

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

SUBJECT: ISSUANCE OF AMENDMENT FOR VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. M96304)

Dear Mr. Reid:

The Commission has issued the enclosed Amendment No. 150 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated August 9, 1996, as supplemented by your letter of September 17, 1996.

The amendment revises the Technical Specifications to revise the safety limit minimum critical power ratio for cycle 19 operation from its current value of 1.07 (for the fuel currently in the reactor for cycle 18) for two recirculation loop operation to 1.10, and from 1.08 to 1.12 for single recirculation loop operation.

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,

C. C. Harbuck

C. Craig Harbuck, Acting Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 150 to DPR-28
2. Safety Evaluation

cc w/encls: See next page

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/S/

Craig Harbuck, Senior Project Manager
Project Directorate I-1
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DOCUMENT NAME: G:\VERMONT\VY96304.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

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DATE	09/19/96	09/19/96	10/1/96	10/4/96	

OFFICIAL RECORD COPY

DATE: October 4, 1996

ISSUANCE OF AMENDMENT NO.150 TO FACILITY OPERATING LICENSE NO. DPR-28

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Vermont Yankee Nuclear Power Station

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated August 9, 1996, as supplemented by letter on September 17, 1996, 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. 150, 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

S. Singh Bajwa

S. Singh Bajwa, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 4, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 150

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
6	6
12	12
13	13
142	142
227	227

1.1 SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and the core flow is greater than 10% of rated:

1. For the Cycle 19 core loading, the existence of a Minimum Critical Power Ratio (MCPR) of less than 1.10 (1.12 for Single Loop Operation) shall constitute violation of the Fuel Cladding Integrity Safety Limit (FCISL). Core loadings subsequent to Cycle 19 will require recalculation of the MCPR.

2.1 LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip setting of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

When the mode switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

$$S \leq 0.66(W-\Delta W)+54\%$$

where:

S = setting in percent of rated thermal power (1593 MWt)

W = percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow

BASES:

1.1 FUEL CLADDING INTEGRITY

A. Refer to General Electric Company Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (most recent revision).

The fuel cladding integrity Safety Limit (SL) is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations.

The MCPR fuel cladding integrity SL is increased for single loop operation in order to account for increased core flow measurement and TIP reading uncertainties.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia or Core Flow ≤10% of Rated)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux

BASES: 1.1 (Cont'd)

following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Vermont Yankee has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the enriched fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the enriched fuel provides adequate margin. This level will be continuously monitored.

BASES: 3.6 and 4.6 (Cont'd)

impurities will also be within their normal ranges. The reactor cooling samples will also be used to determine the chlorides. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride ion content. Isotopic analyses required by Specification 4.6.B.2 may be performed by a gamma scan and gross beta and alpha determination.

The conductivity of the feedwater is continuously monitored and alarm set points consistent with Regulatory requirements given in Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," have been determined. The results from the conductivity monitors on the feedwater can be correlated with the results from the conductivity monitors on the reactor coolant water to indicate demineralizer breakthrough and subsequent conductivity levels in the reactor vessel water.

C. Coolant Leakage

The 5 gpm limit for unidentified leaks was established assuming such leakage was coming from the reactor coolant system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. These tests suggest that for leakage somewhat 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 shutdown 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 equivalent 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 maintaining the fuel cladding integrity safety limit 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.

BASES:

3.11 FUEL RODS

A. Average Planar Linear Heat Generation Rule (APLHGR)

Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analyses methods.

(Note: All exposure increments in this Technical Specification section are expressed in terms of megawatt-days per short ton.)

The MAPLHGR reduction factor of 0.83 for single recirculation loop operation is based on the assumption that the coastdown flow from the unbroken recirculation loop would not be available during a postulated large break in the active recirculation loop, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation." February 1983.

B. Linear Heat Generation Rate (LHGR)

Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analyses methods.

C. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

1. The MCPR operating limit is a cycle-dependent parameter which can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provide reasonable assurance against exceeding the Fuel Cladding Integrity Safety Limit (FCISL) for potential abnormal occurrences. The MCPR operating limits are justified by the analyses, the results of which are presented in the current cycle's Core Performance Analysis Report. Refer to the appropriate topical reports listed in Specification 6.7.A.4 for analysis methods. The increase in MCPR operating limits for single loop operation accounts for increased core flow measurement and TIP reading uncertainties.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated August 9, 1996, as supplemented by your letter of September 17, 1996, the Vermont Yankee Nuclear Power Corporation (the licensee) submitted a request for changes to the Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications (TSs). The proposed changes would revise the safety limit minimum critical power ratio (SLMCPR) for cycle 19 operation from its current value of 1.07 (for the fuel currently in the reactor for cycle 18) for two recirculation loop operation to 1.10, and from 1.08 to 1.12 for single recirculation loop operation. The changes are based on a cycle-specific analyses performed by General Electric (GE). The September 17, 1996, letter provided clarifying information that did not change the scope of the August 9, 1996, application and initial proposed no significant hazards consideration determination.

2.0 EVALUATION

As a result of recent calculations, GE, the source for the Vermont Yankee (VY) fuel and associated safety evaluations, has determined that plant and cycle specific calculations of the SLMCPR may be more conservative and indicate an increased magnitude for the safety limit is required compared to that indicated by the fuel type generic calculations which have formed the bases for previous SLMCPR calculations for VY and other reactors (Reference 1). It is, therefore, necessary that cycle specific calculations be factored into the determinations of SLMCPR for each cycle. These calculations have been done (by GE) for VY for cycle 18 and 19. The calculations show that the SLMCPR values for cycle 19 fuel and core configurations should be increased as indicated above. The calculations for cycle 18 indicate that the current values are acceptable for the remainder of cycle 18.

The GE calculations were done with the methodologies and procedures described in the NRC-approved GE report GESTAR II, Revision 11 (Reference 2), Sections 1.1.5 and 1.2.5, but using parameters specific for the VY fuel and core configurations of cycles 18 and 19. The NRC staff review concludes that this approach is acceptable while the staff is still working to resolve the generic procedures used for bounding SLMCPR calculations. The proposed values for the SLMCPR are acceptable because they will ensure at least 99.9% of the fuel rods in the core do not experience transition boiling and no significant radiological release will result during the most limiting analyzed transient. The approval for the TS change applies only for cycle 19 and the remainder of cycle 18.

Specification 1.1.A, Safety Limit, has been changed to specify only one set of safety limits with values of 1.10 and 1.12 for single and two loop operation. The TS also specifies that these values are acceptable for cycle 19 only and that subsequent loadings will require recalculation of MCPR. These changes are consistent with the analyses and are acceptable. Future cycles will be considered following conclusion of the staff generic review.

Bases 1.1.A and D have been changed to reflect the TS changes. These changes provide an acceptable statement of the staff approved SLMCPR changes and are acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 44364). 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.

6.0 REFERENCES

1. Letter from M. A. Smith, G.E., to Document Control Desk, 10 CFR Part 21, Reportable Condition, Safety Limit MCPR Evaluations, May 24, 1996.

2. GESTAR II, NEDE-24011, Revision 11-P-A, November 17, 1995.

Principal Contributor: H. Richings

Date: October 4, 1996