December 4, 1984

Docket No. 50-271

Mr. R. W. Capstick
Licensing Engineer
Vermont Yankee Nuclear Power Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Capstick:

The Commission has issued the enclosed Amendment No. 84 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications in response to your application dated January 23, 1984.

This amendment revises the Technical Specifications to reflect a change from 850 to 800 psig in the main steam line low pressure isolation setpoint.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Vernon L. Rooney, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 84 to
   License No. DPR-28
2. Safety Evaluation

cc w/enclosure:
See next page

DISTRIBUTION	EMarinos		
Docket File	SNorris	PMcKee	OPA, CMiles
NRC PDR	VRooney	TBarnhart (4)	RDiggs
Local PDR	OELD	WJones	Gray File
ORB#2 Reading	LJHarmon	DBrinkman	Extra - 5
DEisenhut	ELJordan	ACRS (10)	JPartlow

DL:ORB#2	DL:ORB#2	DL:ORB#2	OELD	DL:AD-OR
SNorris:ajs	VRooney	DVassallo	Bachmann	GLainas
11/27/84	11/27/84	11/27/84	11/29/84	12/4/84

Mr. R. W. Capstick
Vermont Yankee Nuclear Power Corporation
Vermont Yankee Nuclear Power Station

cc:

Mr. W. F. Conway
President & Chief Executive Officer
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Donald Hunter, Vice President
Vermont Yankee Nuclear Power Corp.
1671 Worcester Road
Framingham, Massachusetts 01701

New England Coalition on
Nuclear Pollution
Hill and Dale Farm
R. D. 2, Box 223
Putney, Vermont 05346

Mr. Walter Zaluzny
Chairman, Board of Selectman
Post Office Box 116
Vernon, Vermont 05345

J. P. Pelletier, Plant Manager
Vermont Yankee Nuclear Power Corp.
Post Office Box 157
Vernon, Vermont 05354

Raymond N. McCandless
Vermont Division of Occupational
& Radiological Health
Administration Building
10 Baldwin Street
Montpelier, Vermont 05602

Honorable John J. Easton
Attorney General
State of Vermont
109 State Street
Montpelier, Vermont 05602

John A. Ritscher, Esquire
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

W. P. Murphy, Vice President &
Manager of Operations
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

U. S. Environmental Protection
Agency
Region I Office
Regional Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

Vermont Yankee Decommissioning
Alliance
Box 53
Montpelier, Vermont 05602-0053

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 176
Vernon, Vermont 05354

Vermont Public Interest
Research Group, Inc.
43 State Street
Montpelier, Vermont 05602

Thomas A. Murley
Regional Administrator
Region I Office
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. Richard Saudek, Commissioner
Vermont Department of Public Service
120 State Street
Montpelier, Vermont 05602

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. 84
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 January 23, 1984 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 2.C.(2) of Facility Operating License No. DPR-28 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 84 , 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 4, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 84

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise the Technical Specifications as follows:

Remove	Insert
3	3
7	7
14a	14a
15	15
15a	15a
41	41
64	64

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1593 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1593 thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser volume trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 800 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

1.1 SAFETY LIMIT

D. Whenever the reactor is shutdown with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the enriched fuel when it is seated in the core.

2.1 LIMITING SAFETY SYSTEM SETTING

C. Reactor low water level scram setting shall be at least 127 inches above the top of the enriched fuel.

D. Reactor low-low water level Emergency Core Cooling System (ECCS) initiation shall be at least 82.5 inches above the top of the enriched fuel.

E. Turbine stop valve scram shall be less than or equal to 10% valve closure from full open.

F. Turbine control valve fast closure scram shall, when operating at greater than 30% of full power, trip upon actuation of the turbine control valve fast closure relay.

G. Main steam line isolation valve closure scram shall be less than or equal to 10% valve closure from full open.

H. Main steam line low pressure initiation of main steam line isolation valve closure shall be at least 800 psig.

APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. If the scram requires a change due to an abnormal peaking condition, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1.A.1.a, thus assuring a reactor scram at lower than design overpower conditions.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR (Specification 3.11C).

Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of the rated. (During an outage when it is necessary to check refuel interlocks, the mode switch must be moved to the startup position. Since the APRM reduced scram may be inoperable at that time due to the disconnection of the LPRMs, it is required that the IRM scram and the SRM scram in noncoincidence be in effect. This will ensure that adequate thermal margin is maintained between the setpoint and the safety limit.) The margin is adequate to accommodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 800 psig.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument, which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale for that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accommodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For in-sequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any safety limit is exceeded.

2.1 (cont.)

D. Reactor Low Water Level ECCS Initiation Trip Point

The core standby cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident and to limit fuel clad temperature to well below the clad melting temperature, and to limit clad metal-water reaction to less than 1%, to assure that core geometry remains intact.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the ECCS initiation setpoint would now prevent the ECCS components from meeting their design criteria. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Closure Scram Trip Setting

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of <10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the fuel cladding integrity safety limit even during the worst case transient that assumes the turbine bypass is closed. This scram is bypassed when turbine steam flow is below 30% of rated, as measured by turbine first stage pressure.

F. Turbine Control Valve Fast Closure Scram

The control valve fast closure scram is provided to limit the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection coincident with failure of the bypass system. This transient is less severe than the turbine stop valve closure with failure of the bypass valves and therefore adequate margin exists.

G. Main Steam Line Isolation Valve Closure Scram

The isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scram setpoint at 10% of valve closure, there is no increase in neutron flux.

H. Reactor Coolant Low Pressure Initiation of Main Steam Isolation Valve Closure

The low pressure isolation of the main steam lines at 800 psig is provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage is taken of the scram feature which occurs when the main steam line isolation valves are closed, to provide the reactor shutdown so that high power operation at low reactor pressure does not

2.1 (cont.)

occur. Operation of the reactor at pressures lower than 800 psig requires that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram.

Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the available of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

TABLE 3.2.2

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Setting	Required Action When Minimum Conditions for Operation are Not Satisfied (Note 2)
2	Low-Low Reactor Vessel Water Level	≥82.5" above the top of enriched fuel	A
2 of 4 in each of 2 channels	High Main Steam Line Area Temperature	≤ 212°F	B
2/steam line	High Main Steam Line Flow	≤120% of rated flow	B
2/(Note 1)	Low Main Steam Line Pressure	≥800 psig	B
2/(Note 6)	High Main Steam Line Flow	≤40% of rated flow	B
2	Low Reactor Vessel Water Level	Same as Reactor Protection System	A
2	High Main Steam Line Radiation (7) (8)	≤3 X Background at rated power (9)	B
2	High Drywell Pressure	Same as Reactor Protection System	A
2/(Note 10)	Condenser Low Vacuum	≥ 12" Hg absolute	A
1	Trip System Logic	--	A

3.2 (Continued)

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure resulting from a control rod drop accident. This instrumentation causes closure of Group 1 valves, the only valves required to close for this accident. With the established setting of 3 times normal background and main steam line isolation valve closure, fission product release is limited so that 10CFR100 limits are not exceeded for the control rod drop accident and 10CFR20 limits are not exceeded for gross fuel failure during reactor operations. With an alarm setting of 1.5 times normal background, the operator is alerted to possible gross fuel failure or abnormal fission product releases from failed fuel due to transient reactor operation.

Pressure instrumentation is provided which trips when main steam line pressure drops below 800 psig. A trip of this instrumentation results in closure of Group 1 isolation valves. In the refuel, shutdown, and startup modes, this trip function is provided when main steam line flow exceeds 40% of rated capacity. This function is provided primarily to provide protection against a pressure regulator malfunction which would cause the control and/or bypass valves to open, resulting in a rapid depressurization and cooldown of the reactor vessel. The 800 psig trip setpoint limits the depressurization such that no excessive vessel thermal stress occurs as a result of a pressure regulator malfunction. This setpoint was selected far enough below normal main steam line pressures to avoid spurious primary containment isolations.

Low condenser vacuum has been added as a trip of the Group 1 isolation valves to prevent release of radioactive gases from the primary coolant through condenser. The setpoint of 12 inches of mercury absolute was selected to provide sufficient margin to assure retention capability in the condenser when gas flow is stopped and sufficient margin below normal operating values.

The HPCI and/or RCIC high flow, steam supply pressure, and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 6 valves. A time delay has been incorporated into the RCIC steam flow trip logic to prevent the system from inadvertently isolating due to pressure spikes which may occur on startup. The trip settings are such that core uncovering is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual channel system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. Permanently installed circuits and equipment may be used to trip instrument channels. In the non-fail safe systems which require energizing the circuitry, tripping an instrument channel may take the form of providing the required relay function by use of permanently installed circuits. This is accomplished in some cases by closing logic circuits with the aid of the permanently installed test jacks or other circuitry which would be installed for this purpose.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 84 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 Introduction

The Vermont Yankee Nuclear Power Corporation (VYNPC/the licensee), by letter dated January 23, 1984 proposed changes to the Technical Specifications to reduce the main steam line isolation valve (MSIV) low pressure setpoint to 800 psig from 850 psig, when the reactor is in run mode. The principal reason for the change is to reduce challenges to the containment isolation system and safety relief valves. Reduction of challenges to the safety relief valves is consistent with the objectives of the NUREG-0737, Item II.K.3.16, "Reduction in the Challenges and Failures of Relief Valves."

The purpose of the main steam line low pressure isolation is to prevent excessive vessel depressurization and cool down in the event of a pressure regulator malfunction. This low pressure isolation is not required for the primary containment and reactor vessel isolation during a steam line break accident. For steam line breaks, isolation signals are generated from high differential pressure across the main steam line flow restrictors.

2.0 Evaluation

The licensee, in order to determine the safety implications of reducing the MSIV low pressure isolation setpoint, submitted an analysis (NEDO-2243-1, "Safety Evaluation of MSIV Low Pressure Turbine Inlet Pressure Setpoint Change for Vermont Yankee Nuclear Power Station," Rev. 1, dated May 1983).

The analysis simulates plant response to a pressure regulator failure (open) assuming a 750 psig setpoint. The assumed 750 psig setpoint bounds the proposed 800 psig setpoint. The calculated maximum change in the vessel steam dome saturation temperature was used to calculate a vessel component peak thermal stress and fatigue usage factor. The calculations indicate that lowering the low pressure isolation setpoint from 850 psig to 800 psig will have a negligible effect on the reactor vessel's lifetime fatigue usage.

In the licensee's limiting transient analysis the vessel steam dome saturation temperature is reduced from 549°F to 510.6°F. At these temperatures, the vessel materials will behave in a ductile rather than brittle fashion. Since the reactor vessel materials will behave in a ductile fashion during the limiting transient, the staff considers that reducing the proposed setpoint to 800 psig, will not increase the risk of brittle fracture of the reactor vessel.

The licensee has also evaluated the radiological releases resulting from the lowering of the MSIV low pressure isolation setpoint. It was determined that for the design basis main steam line break outside containment, the calculated radiological releases will not change since the MSIV isolation is assumed to occur on high steam line flow rate, not on low main steam line pressure. Breaks that are too small to be detected by the high flow sensors are assumed to be detected either by temperature sensors in the steam tunnel or area radiation monitors in the turbine building. We agree with the licensee's conclusion that no change in doses would be calculated for a main steam line break accident as a result of this change.

The maximum critical power ratio (MCPR) limit will also not be affected by the reduced low pressure isolation setpoint because a reactor water level scram will reduce power level before the pressure of 850 psig is reached. Therefore, any subsequent differences between 850 psig and 750 psig will not affect the calculated MCPR limit.

3.0 Summary

Based on the analysis results provided by the licensee, in support of the MSIV low pressure isolation setpoint from 850 psig to 800 psig we have concluded that there will be no adverse effects on 1) the vessel's lifetime fatigue usage, 2) the vessel material's brittle fracture resistance, 3) the radiological releases and 4) the MCPR operating limits. Therefore, we find the proposed Technical Specification change acceptable.

4.0 Environmental Considerations

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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 issued a proposed finding that this 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.

5.0 Conclusion

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

Principal Contributor: E. Marinos

Dated: December 4, 1984