Mr. Samuel L. Newton
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, VT 05301

April 25, 2000

Template NRR-058

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: ENHANCEMENTS TO SUPPORT INCREASED CORE FLOW (TAC NO. MA6152)

Dear Mr. Newton:

The Commission has issued the enclosed Amendment No. 187 to Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station, in response to your application dated July 20, 1999, as supplemented on October 25, 1999. The October 25, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

The amendment revises the Technical Specifications to reflect the implementation of increased core flow. 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,
/RA/

Richard P. Croteau, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures:
1. Amendment No. 187 to License No. DPR-28
2. Safety Evaluation

cc w/encls: See next page

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

April 25, 2000

Mr. Samuel L. Newton
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, VT 05301

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cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. David R. Lewis
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037-1128

Mr. Richard P. Sedano, Commissioner
Vermont Department of Public Service
112 State Street
Montpelier, VT 05620-2601

Mr. Michael H. Dworkin, Chairman
Public Service Board
State of Vermont
112 State Street
Montpelier, VT 05620-2701

Chairman, Board of Selectmen
Town of Vernon
P.O. Box 116
Vernon, VT 05354-0116

Mr. Richard E. McCullough
Operating Experience Coordinator
Vermont Yankee Nuclear Power Station
P.O. Box 157
Governor Hunt Road
Vernon, VT 05354

G. Dana Bisbee, Esq.
Deputy Attorney General
33 Capitol Street
Concord, NH 03301-6937

Chief, Safety Unit
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Deborah B. Katz
Box 83
Shelburne Falls, MA 01370

Mr. Raymond N. McCandless
Vermont Department of Health
Division of Occupational
and Radiological Health
108 Cherry Street
Burlington, VT 05402

Mr. Gautam Sen
Licensing Manager
Vermont Yankee Nuclear Power
Corporation
185 Old Ferry Road
Brattleboro, VT 05301

Resident Inspector
Vermont Yankee Nuclear Power Station
U. S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, VT 05354

Director, Massachusetts Emergency
Management Agency
ATTN: James Muckerheide
400 Worcester Rd.
Framingham, MA 01702-5399

Jonathan M. Block, Esq.
Main Street
P. O. Box 566
Putney, VT 05346-0566

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. 187
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 July 20, 1999, as supplemented on October 25, 1999, 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:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 187 , 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 25, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 187

FACILITY OPERATING LICENSE NO. DPR-28

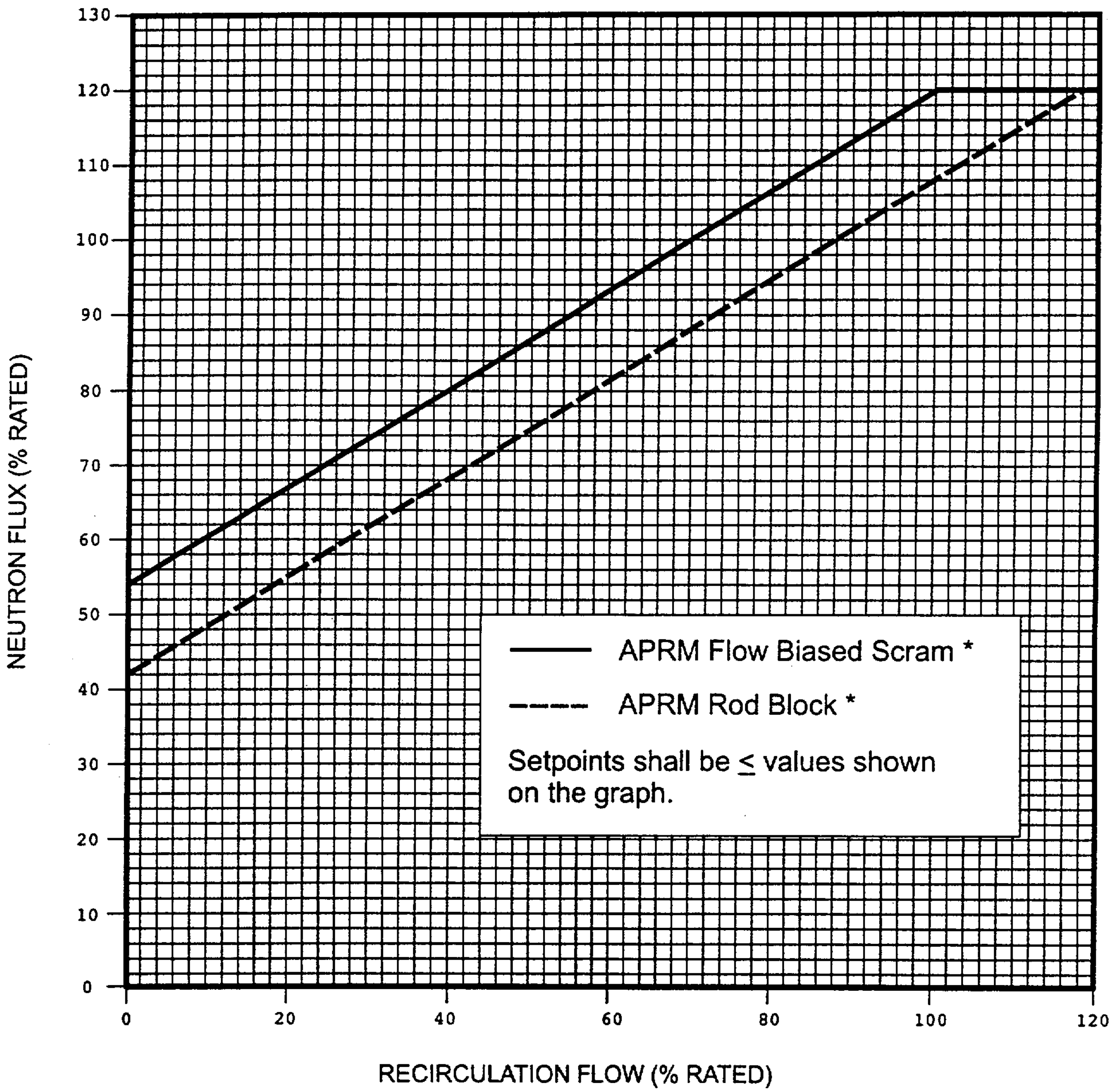
DOCKET NO. 50-271

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove	Insert
11	11
16	16
17	17
21	21
51	51
75	75
76	76
77	77
78	78
79	79

FIGURE 2.1-1

APRM FLOW REFERENCE SCRAM AND APRM ROD BLOCK SETTINGS



* For single loop operation, the APRM Scram and Rod Block settings are adjusted according to Technical Specifications 2.1.A.1.a and 2.1.B.1

BASES: 2.1 (Cont'd)

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case involves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. Additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

B. APRM Rod Block Trip Setting

The purpose of the APRM rod block function is to avoid conditions that would require Reactor Protection System action if allowed to proceed. The APRM upscale rod block alarm setting is selected to initiate a rod block before the APRM high neutron flux scram setting is reached. The APRM upscale rod block trip setpoint is varied as a function of reactor recirculation flow. The slope of the rod block trip response curve with recirculation flow is adjustable to allow tracking of the required trip setpoint with recirculation flow changes. This provides an effective rod block if core average power is increased above the power level specified at any flow rate. As with the APRM flux scram trip setting, the APRM rod block trip setting is reduced for single recirculation loop operation in accordance with the analysis presented in NEDO-30060, February, 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow.

C. Reactor Low Water Level Scram

The reactor low water level scram is set at a point which will prevent reactor operation with the steam separators uncovered, thus limiting carry-under to the recirculation loops. In addition, the safety limit is based on a water level below the scram point and therefore this setting is provided.

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.

BASES: 2.1 (Cont'd)

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 signal may be bypassed at ≤30% of reactor Rated Thermal Power.

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. This scram signal may be bypassed at ≤30% of reactor Rated Thermal Power.

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 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 availability of neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

	Trip Function	Trip Settings	Modes in Which Functions Must be Operating: Refuel (1)	Startup (12)	Run	Minimum Number Operating Instrument Channels Per Trip System (2)	Required ACTIONS When Minimum Conditions For Operation Are Not Satisfied (3)
1.	Mode Switch in Shutdown (5A-S1)		X	X	X	1	A
2.	Manual Scram (5A-S3A/B)		X	X	X	1	A
3.	IRM (7-41(A-F))						
	High Flux	≤120/125	X	X		2	A
	INOP		X	X		2	A
4.	APRM (APRM A-F)						
	High Flux (flow bias)	≤0.66 (W-ΔW)+54% with a maximum of 120% (4)			X	2	A or B
	High Flux (reduced)	≤15%	X	X		2	A
	INOP			X	X	2(5)	A or B
5.	High Reactor Pressure (PT-2-3-55(A-D)(M))	≤1055 psig	X	X	X	2	A
6.	High Drywell Pressure (PT-5-12(A-D)(M))	≤2.5 psig	X	X	X	2	A
7.	Reactor Low (6) Water Level (LT-2-3-57A/B(M)) (LT-2-3-58A/B(M))	≥127.0 inches	X	X	X	2	A
8.	Scram Discharge Volume High Level (LT-3-231(A-H)(M))	≤21 gallons	X	X	X	2 (per volume)	A

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System		Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
		Startup Range Monitor				
	2	a. Upscale (Note 2)(7-40(A-D))	X	X		$\leq 5 \times 10^5$ cps (Note 3)
	2	b. Detector Not Fully Inserted (7-11(A-D)(LS-4))	X	X		
		Intermediate Range Monitor				
(Notes 10, 1)	2	a. Upscale (7-41(A-F))	X	X		≤108/125 Full Scale
	2	b. Downscale (Note 4) (7-41(A-F))	X	X		≥5/125 Full Scale
	2	c. Detector Not Fully Inserted (7-11(E,F,G,H,J,K)(LS-4))	X	X		
		Average Power Range Monitor (APRM A-F)				
	2	a. Upscale (Flow Bias)			X	≤0.66(W-ΔW)+42% (Note 5)
	2	b. Downscale			X	≥2/125 Full Scale
		Rod Block Monitor (RBM A/B)				
(Notes 10, 9)	1	a. Upscale (Flow Bias) (Note 7)			X	≤0.66(W-ΔW)+N with a maximum as defined in the COLR (Note 5)
	1	b. Downscale (Note 7)			X	≥2/125 Full Scale
(Notes 10,11)	1 (per volume)	Scram Discharge Volume (LT-3-231A/G (S1))	X	X	X	≤12 Gallons
(Note 8)	1	Trip System Logic	X	X	X	

BASES:

3.2 PROTECTIVE INSTRUMENTATION

In addition to reactor protection instrumentation which initiates a reactor scram, station protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the reactor operator's ability to control, or terminate a single operator error before it results in serious consequences. This set of Specifications provides the limiting conditions of operation for the primary system isolation function and initiation of the core standby cooling and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of any component of such systems even during periods when portions of such systems are out of service for maintenance, testing, or calibration, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system and surveillance instrumentation.

Isolation valves (Note 1) are installed in those lines that penetrate the primary containment and must be isolated during a loss-of-coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.2 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the limits of 10 CFR 100 are not exceeded during an accident. The objective of the low turbine condenser vacuum trip is to minimize the radioactive effluent releases to as low as practical in case of a main condenser failure. Subsequent releases would continue until operator action was taken to isolate the main condenser unless the main steam line isolation valves were closed automatically on low condenser vacuum. The manual bypass is required to permit initial startup of the reactor during low power operation.

The instrumentation which initiates primary system isolation is connected in a dual channel arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is 127" above the top of the enriched fuel. This trip initiates closure of Group 2 and 3 primary containment isolation valves. For a trip setting of 127" above the top of the enriched fuel, the valves will be closed before perforation of the clad occurs even for the maximum break and, therefore, the setting is adequate.

The top of the enriched fuel (351.5" from vessel bottom) is designated as a common reference level for all reactor water level instrumentation. The intent is to minimize the potential for operator confusion which may result from different scale references.

The low-low reactor water level instrumentation is set to trip when reactor water level is 82.5" H_2O indicated on the reactor water level instrumentation above the top of the enriched fuel. This trip initiates closure of the Group 1 primary containment isolation valves and also activates the ECCS and RCIC System and starts the standby diesel generator system. This trip setting level was chosen to be low enough to prevent spurious operation, but high enough to initiate ECCS operation and primary system isolation so that no melting of the fuel cladding will occur, and so that post-accident cooling can be accomplished and the limits of 10CFR100 will not be violated.

Note 1 - Isolation valves are grouped as listed in Table 3.7.1.

BASES: 3.2 (Cont'd)

For the complete circumferential break of 28-inch recirculation line and with the trip setting given above, ECCS initiation and primary system isolation are initiated in time to meet the above criteria. The instrumentation also covers the full range of spectrum breaks and meets the above criteria.

The high drywell pressure instrumentation is a backup to the water level instrumentation, and in addition to initiating ECCS, it causes isolation of Group 2, 3, and 4 isolation valves. For the complete circumferential break discussed above, this instrumentation will initiate ECCS operation at about the same time as the low-low water level instrumentation, thus, the results given above are applicable here also. Group 2 isolation valves include the drywell vent, purge, and sump isolation valves. High drywell pressure activates only these valves because high drywell pressure could occur as the result of nonsafety-related causes such as not purging the drywell air during startup. Total system isolation is not desirable for these conditions and only the valves in Group 2 are required to close. The water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes a trip of all primary system isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. In addition to monitoring steam flow, instrumentation is provided which causes a trip of Group 1 isolation valves. The primary function of the instrumentation is to detect a break in the main steam line, thus only Group 1 valves are closed. For the worst case accident, main steam line break outside the drywell, this trip setting of 140 percent of rated steam flow in conjunction with the flow limiters and main steam line valve closure limit the mass inventory loss such that fuel is not uncovered, cladding temperatures remain less than 1295°F and release of radioactivity to the environs is well below 10CFR100.

Temperature monitoring instrumentation is provided in the main steam line tunnel to detect leaks in this area. Trips are provided on this instrumentation and when exceeded cause closure of Group 1 isolation valves. Its setting of ambient plus 95°F is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, it is a backup to high steam flow instrumentation discussed above, and for small breaks, with the resultant small release of radioactivity, gives isolation before the limits of 10CFR100 are exceeded.

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. 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

BASES: 3.2 (Cont'd)

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. Permanently installed circuits and equipment may be used to trip instrument channels. In the nonfail 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.

The trip logic for the nuclear instrumentation control rod block logic is 1 out of n; i.e., any trip on one of the six APRMs, six IRMs or four SRMs will result in a rod block. The minimum instrument channel requirements for the IRMs may be reduced by one for a short period of time to allow for maintenance, testing or calibration. The RBM is credited in the Continuous Rod Withdrawal During Power Range Operation transient for preventing excessive control rod withdrawal before the fuel cladding integrity safety limit (MCPR) or the fuel rod mechanical overpower limits are exceeded. The RBM upper limit is clamped to provide protection at greater than 100% rated core flow. The clamped value is cycle specific; therefore, it is located in the Core Operating Limits Report.

For single recirculation loop operation, the RBM trip setting is reduced in accordance with the analysis presented in NEDO-30060, February 1983. This adjustment accounts for the difference between the single loop and two loop drive flow at the same core flow, and ensures that the margin of safety is not reduced during single loop operation.

The purpose of the APRM rod block function is to avoid conditions that would require Reactor Protection System action if allowed to proceed. The APRM upscale rod block alarm setting is selected to initiate a rod block before the APRM high neutron flux scram setting is reached. The APRM upscale rod block trip setpoint is varied as a function of reactor recirculation flow. The slope of the rod block trip response curve with recirculation flow is adjustable to allow tracking of the required trip setpoint with

BASES: 3.2 (Cont'd)

recirculation flow changes. This provides an effective rod block if core average power is increased above the power level specified at any flow rate. For single recirculation loop operation, the APRM rod block trip setting is reduced in accordance with the analysis presented in NEDE-30060, February 1983. This adjustment accounts for the difference between the single loop and two-loop drive flow at the same core flow.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the fuel cladding integrity safety limit.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion and thus control rod motion is prevented.

To prevent excessive clad temperatures for the small pipe break, the HPCI or Automatic Depressurization System must function since, for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. For a break or other event occurring outside the drywell, the Automatic Depressurization System is initiated on low-low reactor water level only after a time delay. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the Specification are adequate to ensure the above criteria are met. The Specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

The ADS is provided with inhibit switches to manually prevent automatic initiation during events where actuation would be undesirable, such as certain ATWS events. The system is also provided with an Appendix R inhibit switch to prevent inadvertent actuation of ADS during a fire which requires evacuation of the Control Room.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation duct and on the refueling floor. Any one upscale trip or two downscale trips of either set of monitors will cause the desired action. Trip settings for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leave the Reactor Building via the normal ventilation stack but that all activity is processed by the standby gas treatment system. Trip settings for the monitors in the ventilation duct are based upon initiation of the normal ventilation isolation and standby gas treatment system operation at a radiation level equivalent to the maximum site boundary dose rate of 500 mrem/year as given in Specification 3.8.E.1.a. The monitoring system in the plant stack represents a backup to this system to limit gross radioactivity releases to the environs.

The purpose of isolating the mechanical vacuum pump line is to limit release of radioactivity from the main condenser. During an accident, fission products would be transported from the reactor through the main steam line to the main condenser. The fission product radioactivity would be sensed by the main steam line radiation monitors which initiate isolation.

BASES: 3.2 (Cont'd)

Post-accident instrumentation parameters for Containment Pressure, Torus Water Level, Containment Hydrogen/Oxygen Monitor, and Containment High-Range Radiation Monitor, are redundant, environmentally and seismically qualified instruments provided to enhance the operators' ability to follow the course of an event. The purpose of each of these instruments is to provide detection and measurement capability during and following an accident as required by NUREG-0737 by ensuring continuous on-scale indication of the following: containment pressure in the (-15) -(+260) psig range; torus water level in the 0 to 25 foot range (i.e., the bottom to 5 feet above the normal water level of the torus pool); containment hydrogen/oxygen concentrations (0 to 30% hydrogen and 0 to 25% oxygen); and containment radiation in the 1 R/hr to 10^7 R/hr gamma.

The Degraded Grid Protective System has been installed to assure that safety-related electrical equipment will not be subjected to sustained degraded voltage. This system incorporates voltage relays on 4160 Volt Emergency Buses 3 and 4 which are set to actuate at the minimum voltage required to prevent damage of safety-related equipment.

If Degraded Grid conditions exist for 10 seconds, either relay will actuate an alarm to alert operators of this condition. Based upon an assessment of these conditions the operator may choose to manually disconnect the off-site power. In addition, if an ESF signal is initiated in conjunction with low voltage below the relay setpoint for 10 seconds, the off-site power will be automatically disconnected.

The Reactor Core Isolation Cooling (RCIC) System provides makeup water to the reactor vessel during shutdown and isolation to supplement or replace the normal makeup sources without the use of the Emergency Core Cooling Systems. The RCIC System is initiated automatically upon receipt of a reactor vessel low-low water level signal. Reactor vessel high water level signal results in shutdown of the RCIC System. However, the system will restart on a subsequent reactor vessel low-low water level signal. The RCIC System is normally lined up to take suction from the condensate storage tank. Suction will automatically switch over from the condensate storage tank to the suppression pool on low condensate storage tank level.

Upon receipt of a LOCA initiation signal, if normal AC power is available, all RHR pumps and both Core Spray pumps start simultaneously with no intentional time delay. If normal AC power is not available, RHR pumps A and D start immediately on restoration of power, RHR pumps B and C start within 3 to 5 seconds of restoration of power and both Core Spray pumps start within 8 to 10 seconds of restoration of power. The purpose of these time delays is to stagger the start of the RHR and Core Spray pumps on the associated Division 1 and Division 2 Buses, thus limiting the starting transients on the 4.16 kV emergency buses. The time delay functions are only necessary when power is being supplied from the standby power sources (EDGs). The time delays remain in the pump start logic at all times as the time delay relay contact is in parallel with the Auxiliary Power Monitor relay contact. Either contact closure will initiate pump start. Thus, the time delays do not affect low pressure ECCS pump operation with normal AC power available. With normal AC power not available, the pump start relays which would have started the B and C RHR pumps and both Core Spray pumps are blocked by the Auxiliary Power Monitor contacts and the pump start time delay relays become the controlling devices.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.187 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 July 20, 1999, as supplemented on October 25, 1999, the Vermont Yankee Nuclear Power Corporation (the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station (VY) Technical Specifications (TSs). The amendment would adjust and cap the flow-biased trip settings in the TSs for the rod block monitor (RBM), the flow-biased average power range monitor (APRM) scram, and the APRM rod block to reflect increased core flow (ICF) operation.

In its application, the licensee stated that it had implemented ICF operation at VY as a design change that was evaluated and implemented per 10 CFR 50.59 after performing mechanical, thermal-hydraulic, and reactor physics evaluations to support operating VY with an ICF of 107 percent. In addition, General Electric analyzed the impact of increased core flow operation on VY's reactor core, recirculation system, and reactor internals. The Cycle 20 steady-state core thermal-hydraulic analysis also incorporated the increased core flow operation in the reload analysis. The licensee determined that the flow-biased RBM, the APRM scram, and the APRM rod block TSs limiting conditions for operations (LCOs) needed to be updated to reflect ICF operation.

ICF is specified as part of the operating flexibility in the NRC-approved General Electric Standard Application for Reactor Fuel (GESTAR II). A licensee can adopt ICF operation under 10 CFR 50.59 without prior NRC approval if all the required analyses support operation with ICF, and the analyses are in compliance with the NRC-approved methodologies specified in the latest revision of GESTAR II. The licensee stated that ICF would allow greater flexibility by eliminating the need to adjust control rods when the control rod patterns are changed and during startups. Core flow changes will compensate for the effects of xenon and fuel burnup. ICF can also be used for cycle extension during end-of-cycle (EOC) when all rods are out.

TSs Tables 3.2.5 and 3.1.1, Figure 2.1-1, and the corresponding Bases sections of the TSs specify the required flow-biased trip settings for the RBM, the flow-biased APRM scram, and the APRM rod block. The flow-biased APRM scram prevents exceeding the thermal-mechanical limits in both rated and off-rated conditions. The flow-biased APRM rod block serves as an aid to the operators. It also prevents the core thermal power from being increased above the design levels by inhibiting control rod withdrawals (which may result in high and rapid power peaking). The RBM is assumed to operate in the final safety analysis report's (FSAR's) analysis of Continuous Rod Withdrawal Error during operation in the power range.

The October 25, 1999, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the original Federal Register notice.

2.0 EVALUATION

The licensee proposed to adjust and cap the flow-biased trip settings for the RBM, APRM scram, and the APRM rod block to reflect operation of VY with an ICF of 107 percent. The following sections describe the licensee's proposed changes, the justifications, and the staff's evaluation:

A. Addition of Clarifying Statement in Figure 2.1-1

In Figure 2.1-1, "APRM Reference Scram and APRM Rod Block Settings," the licensee proposed adding the statement "Setpoints shall be ≤ values shown on the graph." TS Section 2.1 states that the "settings shall be as shown on graph 2.1.1" and the licensee indicated that the statement could be interpreted to mean that the trip setting should be strictly the values specified for the flow-biased APRM scram and the APRM rod block plot. However, the plot is just a graphical representation of the flow biased APRM scram and rod block equations and the settings are required to be less than or equal to the value represented on the plot.

The APRM flow biased scram and rod block trip settings are specified by the linear equation specified in section 3.B of this evaluation and figure 2.1-1 is only a plot of the trip setpoint equations. The staff finds the proposed change to be acceptable because the plot is a graphical representation of the flow biased APRM scram and rod block equations and the settings are required to be less than or equal to the value represented on the plot.

B. Defining the Maximum APRM Flow Biased Trip Setpoint.

Table 3.1.1, "Reactor Protection System (Scram) Instrument Requirements," specifies the trip function settings and item 4 of this table specifies the equation defining the APRM flow-biased upscale trip as $\leq 0.66 (W - \Delta W) + 54$ percent. TS 2.1.A.1.a states: "For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120 percent of rated thermal power." With an increased core flow of 107 percent, the upscale trip setting equation will yield a higher value than the maximum cap of 120 percent thermal power. The licensee proposed adding "with a maximum of 120 percent" to item 4 of Table 3.1.1.

The staff finds the proposed change prudent and acceptable because it limits the maximum APRM flow-biased upscale trip setting in TS Table 3.1.1 such that it does not conflict with the existing requirements in TS 2.1.A.1.a.

C. Rod Block Monitor Flow-Biased Upscale Trip Setting

TS Table 3.2.5, " Control Rod Block Instrumentation," defines the flow-biased upscale and downscale trip settings for the rod block monitor. The VY accident analysis takes credit for the upscale trip setting in the mitigation of the "continuous rod withdrawal transient during power range operation" transient. The TS specifies that the RBM upscale trip setting shall be ≤ 0.66 $(W- \Delta W) +N$, where W is the percent-rated two loop drive flow. The expression ΔW is the difference in the two-loop and single-loop drive flow for the same core flow and must be accounted for during single-loop operation. The variable N is determined in each reload analysis and specified in the core operating limit report (COLR.) The licensee proposes to add the phrase "with a maximum as defined in the COLR" to Table 3.2.5 under the RBM flow-biased upscale trip setting. The licensee states that "since the variable 'N' value in the equation is cycle-specific and is a variable defined in the COLR, the clamped value will also be cycle-specific and it is defined in the COLR."

The staff agrees with the licensee that variable N is cycle-specific and defined in the COLR, so that the RBM flow-biased upscale trip setting is also cycle-specific. The COLR also specifies the maximum upscale flow-biased trip setting. The phrase to be added clarifies Table 3.2.5 and is more restrictive than the current TS. The staff finds the proposed change acceptable because it ensures that the associated assumptions of the "continuous rod withdrawal during power range operation" transient are met.

D. Bases Section 2.1B and Section 3.2

The licensee proposed to change the APRM flow-biased rod block trip setting discussion in Bases Sections 2.1.B (Page 16) and 3.2 (Page 77). The licensee stated that the "current discussion of the APRM Rod Block Trip Setting is ambiguous" and "it implies that the APRM Rod Block Trip setting is assumed in the protection of the fuel integrity safety limit." The licensee asserted that the APRM rod block trip setting is not assumed in any accident or transient analysis; rather, the trip setting stops the reactivity addition before the scram setpoint is reached. The licensee proposed to change the Bases to reflect verbatim the discussions in FSAR Sections 7.7.4.5.2 and 7.5.7.3. According to the licensee, this will clarify Bases 2.1.B and 3.2. The licensee concluded that the changes are administrative in nature since they only clarify the TS Bases, by incorporating information from the FSAR.

The licensee proposed replacing a paragraph on the rod block monitor (Insert 2) and the APRM rod block trip setting (Insert 3) on page 77 of Bases Section 3.2. The staff finds insert 2 an improvement on the RBM discussion in the current Basis.

The staff does not object 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 comment.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement 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 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 (64 FR 46450). 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: Z. Abdullahi

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