August 8, 1986

MEMORANDUM FOR: Sholly Coordinator

FROM: Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

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SUBJECT: REQUEST FOR PUBLICATION IN BI-WEEKLY FR NOTICE - NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of application for amendment: March 12, 1986, March 27, 1986 and May 9, 1986

Brief description of amendment: The amendment revises the Technical Specifications to permit extended reactor operation with one recirculation loop out of service.

Date of issuance: August 8, 1986

Effective date: August 8, 1986

Amendment No.: 94

Facility Operating License No. DPR-28: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 21, 1986 (51 FR 18696)

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated August 8, 1986

No significant hazards consideration comments received: No

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301.

Original signed by
Daniel R. Muller

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PDR ADOCK 05000271
P PDR

Daniel R. Muller, Director
BWR Project Directorate #2
Division of BWR Licensing

DBL:PD2	DBL:PD2	OGC	DBL:PD2
SNorris	VRooney:ps	M. Karman	DMuller
07/22/86	07/23/86	07/24/86	08/8/86

Docket No.: 50-271

August 8, 1986

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. 94 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 March 12, 1986 as revised by letter dated May 9, 1986, with supplemental information provided by letter dated March 27, 1986.

The amendment revises the Technical Specifications to permit extended reactor operation with one recirculation loop out of service.

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

Sincerely,

Original signed by

Vernon L. Rooney, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 94 to License No. DPR-28
2. Safety Evaluation

cc w/enclosure:
See next page

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P PDR

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

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated March 12, 1986, as supplemented March 27 and May 9, 1986, 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) <u>Technical Specifications</u>

The Technical Specifications, contained in Appendix A, as revised through Amendment No. 94 , 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

Daniel R Muller

Daniel R. Muller, Project Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 8, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise the Appendix A Technical Specifications as indicated below. The revised areas are indicated by margional lines.

Pages Deleted	Pages Inserted
5	5
5-a	5-a
5-b	5-b
-	5-c
6	6
6-a	6-a
8	8
9	9
14-a	14-a
14-b	14-b
14-c	14-c
19	19
21	21
47	47
48	48
64-a	64-a
65	65
110	110
110-a	110-a
110-b	110-b
110-c	110-c
110-d	110-d
110-e	110-e
110-f	110-f
-	110-g
-	110-h
-	110-i
-	110-j
-	111-c
124	124
-	124-a
-	124-b
125	125
180-a	180-a
180-c	180-c
180-d	180-d
180-h	180-h
180-n	180-n
180-n1	180-n1
180-n2	180-n2
180-n3	180-n3
180-n5	180-n5
180-01	180-01

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 core flow is >10% of rated, the existence of a Minimum Critical Power Ratio (MCPR) less than 1.07 (1.08 for single loop operation) shall constitute violation of the fuel cladding integrity safety limit.

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:

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

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

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

ΔW = difference between two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two loop operation.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio: $\frac{MFLPD}{FRP}$

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

where:

MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 kW/ft for 8 x 8 fuel.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

b. <u>Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)</u>

When the reactor mode switch is in the REFUEL or STARTUP position, average power range monitor (APRM) scram shall be set down to less than or equal to 15% of rated neutron flux (except as allowed by Note 12 of Table 3.1.1). The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

1.1 SAFETY LIMIT

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

When the reactor pressure is $\leq$800 psia or core flow $\leq$10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

To ensure that the safety limit established in Specification 1.1A and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The safety limit shall be assumed to be exceeded when scram is accomplished by means other than the expected scram signal.

2.1 LIMITING SAFETY SYSTEM SETTING

B. APRM Rod Block Trip Setting

1. The APRM rod block trip setting shall be as shown in Figure 2.1.1 and shall be:

$$S_{RB} \leq 0.66(W-\Delta W) + 42\%$$

where:

S_{RB} = rod block 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

ΔW = difference between two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. ΔW = 0 for two loop operation.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

APRM gain shall be increased by the ratio: $\frac{\text{MFLPD}}{\text{FRP}}$

where:

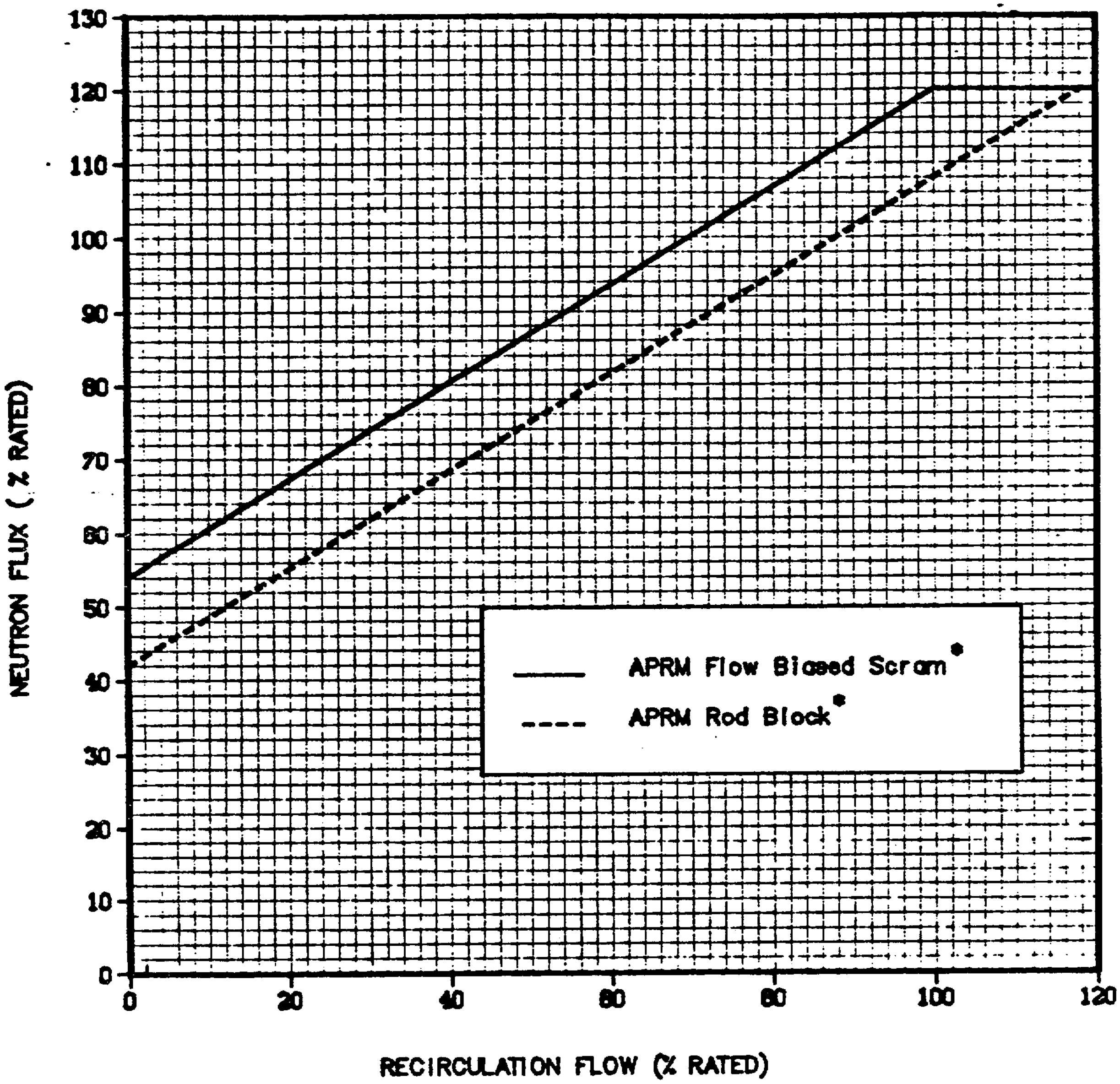
MFLPD = maximum fraction of limiting power density where the limiting power density is 13.4 kW/ft for 8 x 8 fuel.

FRP = fraction of rated power (1593 MWt).

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

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:

1.1 Fuel Cladding Integrity

Refer to Section S.2 of General Electric Company Licensing Topical Report, "United States Supplement, General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A-US (Most Recent Revision).

The MCPR fuel cladding integrity safety limit is increased by 0.01 for single loop operation in order to account for increased core flow measurement and TIP reading uncertainties, as discussed in "Vermont Yankee Nuclear Power Station Single Loop Operation", NEDO-30060, February 1983.

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. For single recirculation loop operation, the APRM flux scram 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.

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.

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

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at the fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship, therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting must be adjusted downward if the ratio of MFLPD to FRP exceeds the specified value. If the APRM rod block requires a change due to abnormal peaking conditions, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1B, thus ensuring a rod block at lower than design overpower conditions. 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, and ensures that the margin of safety is not reduced during single loop operation.

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.

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 Conditions When Minimum Conditions For Operation Are Not Satisfied (3)
1. Mode Switch in Shutdown		X	X	X	1	A
2. Manual Scram		X	X	X	1	A
3. IRM						
High Flux	≤120/125	X	X	X(11)	2	A
INOP		X	X	X(11)	2	A
4. APRM						
High Flux (flow bias)	≤0.66 (W-ΔW)+54% (4)			X	2	A or B
High Flux (reduced)	≤15%	X	X		2	A
INOP				X	2(5)	A or B
Downscale	≥2/125			X	2	A or B
5. High Reactor Pressure	≤1055 psig	X	X	X	2	A
6. High Drywell Pressure	≤2.5 psig	X	X	X	2	A
7. Reactor Low (6) Water Level	≥127.0 inches	X	X	X	2	A
8. Scram Discharge Volume High Level	≤21 gallons	X	X	X	2 (per volume)	A

TABLE 3.1.1 NOTES

1. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:

   a) mode switch in shutdown
   b) manual scram
   c) high flux IRM or high flux SRM in coincidence
   d) scram discharge volume high water level

2. Whenever an instrument system is found to be inoperable, the instrument system output relay shall be tripped immediately. Except for MSIV and Turbine Stop Valve Position, this action shall result in tripping the trip system.

3. When the requirements in the column "Minimum Number of Operating Instrument Channels Per Trip System" cannot be met for one system, that system shall be tripped. If the requirements cannot be met for both trip systems, the appropriate actions listed below shall be taken:

   a) Initiate insertion of operable rods and complete insertion of all operable rods within four hours.
   b) Reduce power level to IRM range and place mode switch in the "Startup/Hot Standby" position within eight hours.
   c) Reduce turbine lead and close main steam line isolation valves within 8 hours.
   d) Reduce reactor power to less than 30% of rated within 8 hours.

4. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

5. To be considered operable an APRM must have at least 2 LPRM inputs per level and at least a total of 13 LPRM inputs, except that channels A, C, D, and F may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

6. The top of the enriched fuel has been designated as 0 inches and provides common reference level for all vessel water level instrumentation.

7. Channel shared by the Reactor Protection and Primary Containment Isolation Systems.

8. An alarm setting of 1.5 times normal background at rated power shall be established to alert the operator to abnormal radiation levels in primary coolant.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System (Note 1)	Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
	Startup Range Monitor				
(Note 1) 2	a. Upscale (Note 2)	X	X		$\leq 5 \times 10^5$ cps (Note 3)
2	b. Detector Not Fully Inserted	X	X		
	Intermediate Range Monitor				
2	a. Upscale	X	X		$\leq$108/125 Full Scale
2	b. Downscale (Note 4)	X	X		$\geq$5/125 Full Scale
2	c. Detector Not Fully Inserted	X	X		
	Average Power Range Monitor				
2	a. Upscale (Flow Bias)			X	$\leq 0.66(W-\Delta W)+42\%$ (Note 5)
2	b. Downscale			X	$\geq$2/125 Full Scale
(Note 9)	Rod Block Monitor (Note 6)				
1	a. Upscale (Flow Bias)(Note 7)			X	$\leq 0.66(W-\Delta W)+N$ (Note 5)
1	b. Downscale (Note 7)			X	$\geq$ 2/125 Full Scale
(Note 8) 1 (per volume)	Scram Discharge Volume	X	X	X	$\leq$12 Gallons
1	Trip System Logic	X	X	X	

TABLE 3.2.5 NOTES

1. There shall be two operable or tripped trip systems for each function in the required operating mode. If the minimum number of operable instruments are not available for one of the two trip systems, this condition may exist for up to seven days provided that during the time the operable system is functionally tested immediately and daily thereafter; if the condition lasts longer than seven days, the system shall be tripped. If the minimum number of instrument channels are not available for both trip systems, the systems shall be tripped.

2. One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

3. This function may be bypassed when count rate is $\geq$100 cps or when all IRM range switches are above Position 2.

4. IRM downscale may be bypassed when it is on its lowest scale.

5. "W" is percent rated two loop drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. Refer to L.C.O. 3.11.C for acceptable values for N. ΔW is the difference between the two loop and single loop drive flow at the same core flow. This difference must be accounted for during single loop operation. $\Delta W = 0$ for two recirculation loop operation.

6. The minimum number of operable instrument channels may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30-day period.

7. The trip may be bypassed when the reactor power is $\leq$30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

8. With the number of operable channels less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

9. With one RBM channel inoperable:

   a. Verify that the reactor is not operating on a limiting control rod pattern, and

   b. Restore the inoperable RBM channel to operable status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

3.2 (Continued)

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease below the fuel cladding integrity safety limit. The trip logic for this function is 1 out of n; e.g., 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 IRM may be reduced by one for a short period of time to allow for maintenance, testing or calibration. The RBM is an operational guide and aid only and is not needed for rod withdrawal.

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.

3.2 (Continued)

The APRM rod block trip is flow referenced and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than the fuel cladding integrity safety limit. For single recirculation loop operation, the APRM rod block 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 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. 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 assure 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.

3.6 LIMITING CONDITION FOR OPERATION

G. Single Loop Operation

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit and MCPR operating limits (Specifications 1.1.A and 3.11.C), and MAPLHGR limits (Specification 3.11.A) are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

   b. With one recirculation pump not in operation, core thermal power greater than the limit specified in Figure 3.6.4, and core flow between 34% and 45% of rated (Region 2 of Figure 3.6.4):

4.6 SURVEILLANCE REQUIREMENT

3. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle.

G. Single Loop Operation

1. With one recirculation pump not in operation, core flow between 34% and 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Region 2), establish baseline APRM and LPRM(1) neutron flux noise levels prior to entering this region, provided that baseline values have not been established since the last core refueling. Baseline values shall be established with one recirculation pump not in operation and core thermal power less than or equal to the limit specified in Figure 3.6.4.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITION FOR OPERATION

4.6 SURVEILLANCE REQUIREMENT

(1) If baseline APRM and LPRM[1] neutron flux noise levels have been established since the last core refueling, initiate action within 15 minutes such that the APRM and LPRM[1] neutron flux noise levels are determined within 2 hours, and:

(i) If the APRM and LPRM[1] neutron flux noise levels are less than or equal to 3 times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a core thermal power increase greater than 5% of rated core thermal power, or

(ii) If the APRM and/or LPRM[1] neutron flux noise levels are greater than 3 times their established baseline levels, initiate action within 15 minutes such that the noise levels are restored to within the required limits within 2 hours by increasing core flow and/or by initiating an orderly reduction of core thermal power by inserting control rods.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITION FOR OPERATION

(2) If baseline APRM and LPRM[1] neutron flux noise levels have not been established since the last core refueling, initiate action within 15 minutes of entering this region (Region 2 of Figure 3.6.4) such that operation is outside this region within 2 hours.

c. With one recirculation pump not in operation, core thermal power greater than the limit specified in Figure 3.6.4, and core flow less than 34% of rated (Region 1 of Figure 3.6.4), initiate action within 15 minutes of entering this region such that operation is outside this region within 2 hours.

d. The idle loop is isolated by electrically disarming the breaker to the recirculation pump motor generator set drive motor prior to startup or, if disabled during reactor operation, within 24 hours, and until such time as the inactive recirculation loop is to be returned to service.

e. The recirculation system controls will be placed in the manual flow control mode.

4.6 SURVEILLANCE REQUIREMENT

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITION FOR OPERATION

H. Recirculation System

1. With two recirculation pumps in operation, with total core flow less than 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Regions 1 and 2):

   a. If baseline APRM and LPRM(1) neutron flux noise levels have been established since the last core refueling, initiate action within 15 minutes such that the APRM and LPRM(1) neutron flux noise levels are determined within 2 hours, and:

      (1) If the APRM and LPRM(1) neutron flux noise levels are less than or equal to 3 times their established baseline levels, continue to determine the noise levels at least once per 8 hours and within 30 minutes after the completion of a core thermal power increase greater than 5% of rated core thermal power, or

4.6 SURVEILLANCE REQUIREMENT

H. Recirculation System

1. With two recirculation pumps in operation, total core flow less than 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Regions 1 and 2), establish baseline APRM and LPRM(1) neutron flux noise levels prior to entering these regions, provided that baseline values have not been established since the last core refueling. Baseline values shall be established with core thermal power less than or equal to the limit specified in Figure 3.6.4.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITION FOR OPERATION

(2) If the APRM and/or LPRM[(1)] neutron flux noise levels are greater than 3 times their established baseline levels, initiate action within 15 minutes such that the noise levels are restored to within the required limits within 2 hours by increasing core flow and/or by initiating an orderly reduction of core thermal power by inserting control rods.

b. If baseline APRM and LPRM[(1)] neutron flux noise levels have not been established since the last core refueling, initiate action within 15 minutes of entering these regions (Regions 1 and 2 of Figure 3.6.4) such that operation is outside these regions within 2 hours.

2. Operation with one recirculation loop is permitted according to Specification 3.6.G.1.

3. With no reactor coolant system recirculation loops in operation, immediately initiate an orderly reduction in core thermal power to less than or equal to the limit specified in Figure 3.6.4 (Region 3), and initiate measures such that the unit is in Hot Shutdown within the next 12 hours.

4.6 SURVEILLANCE REQUIREMENT

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITION FOR OPERATION

I. Shock Suppressors (Snubbers)

1. Except as noted in 3.6.I.2 and 3.6.I.3 below, all required safety-related snubbers shall be operable whenever its supported system is required to be operable.

2. With one or more required snubbers inoperable, within 72 hours, replace or restore the snubber to operable status and perform an engineering evaluation per Specification 4.6.I.1b and c, on the supported component. In all cases, the required snubbers shall be made operable or replaced prior to reactor startup.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, the supported system shall be declared inoperable and the appropriate action statement for that system shall be followed.

4.6 SURVEILLANCE REQUIREMENT

I. Shock Suppressors (Snubbers)

1. Each snubber shall be demonstrated operable by performance of the following inspection program.

   a. Visual Inspections

   Visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Next Required Inspection Intervals
0	18 months ±25%
1	12 months ±25%
2	6 months ±25%
3, 4	124 days ±25%
5, 6, 7	62 days ±25%
8 or more	31 days ±25%

   The snubbers may be categorized into two groups: the accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule. The inspection interval shall not be lengthened more than one step at a time. Inaccessible snubbers are required to be inspected only if the period of time in which they become accessible is greater than 48 hours.

3.6 LIMITING CONDITION FOR OPERATION

4.6 SURVEILLANCE REQUIREMENT

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, and (2) that the snubber installation exhibits no visual indications of detachment from foundations or supporting structures. Snubbers which appear inoperable as a result of visual inspections may be determined operable for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined operable per Specification 4.6.I.c, as applicable. When the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable unless it can be determined operable via functional testing for the purpose of establishing the next visual inspection interval. The functional test, in this case, shall be started with the piston in the as-found condition, extending the piston rod in the tension mode direction.

3.6 LIMITING CONDITION FOR OPERATION

4.6 SURVEILLANCE REQUIREMENT

c. <u>Functional Tests</u>

At least once per 18 months during shutdown, a representative sample of 10% of the snubbers in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.6.I.1.d, an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

Snubbers of a rated capacity greater than the capability of the testing machine shall be functionally tested as follows: (1) the lock up and bleed velocity of the snubber valve shall be verified by testing it on a cylinder that is within the capability of the testing machine, (2) the free stroke of the cylinder shall be checked, and (3) the pressure retaining capability of the cylinder shall be checked.

3.6 LIMITING CONDITION FOR OPERATION

4.6 SURVEILLANCE REQUIREMENT

Snubbers identified as especially difficult to remove or in high radiation areas shall also be included in the representative sample.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period unless the root cause for the problem has been determined and corrective actions implemented. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested during the next test period. Failure of these snubbers shall not entail functional testing of additional snubbers.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency, all

3.6 LIMITING CONDITION FOR OPERATION

G. Single Loop Operation

1. The reactor may be started and operated or operation may continue with a single recirculation loop provided that:

   a. The designated adjustments for APRM flux scram and rod block trip settings (Specifications 2.1.A.1.a and 2.1.B.1, Table 3.1.1 and Table 3.2.5), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit and MCPR operating limits (Specifications 1.1.A and 3.11.C), and MAPLHGR limits (Specification 3.11.A) are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.

   b. With one recirculation pump not in operation, core thermal power greater than the limit specified in Figure 3.6.4, and core flow between 34% and 45% of rated (Region 2 of Figure 3.6.4):

4.6 SURVEILLANCE REQUIREMENT

3. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle.

G. Single Loop Operation

1. With one recirculation pump not in operation, core flow between 34% and 45% of rated, and core thermal power greater than the limit specified in Figure 3.6.4 (Region 2), establish baseline APRM and LPRM[1] neutron flux noise levels prior to entering this region, provided that baseline values have not been established since the last core refueling. Baseline values shall be established with one recirculation pump not in operation and core thermal power less than or equal to the limit specified in Figure 3.6.4.

(1) Detector Levels A and C of one LPRM string per core octant plus detector Levels A and C of one LPRM string in the center of the core shall be monitored.

3.6 LIMITING CONDITION FOR OPERATION

J. Thermal Hydraulic Stability

1. When the reactor mode switch is in RUN, the reactor shall not intentionally be operated in a natural circulation mode, except as permitted by Specification 3.6.H.3, nor shall an idle recirculation pump be started with the reactor in a natural circulation mode.

4.6 SURVEILLANCE REQUIREMENT

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

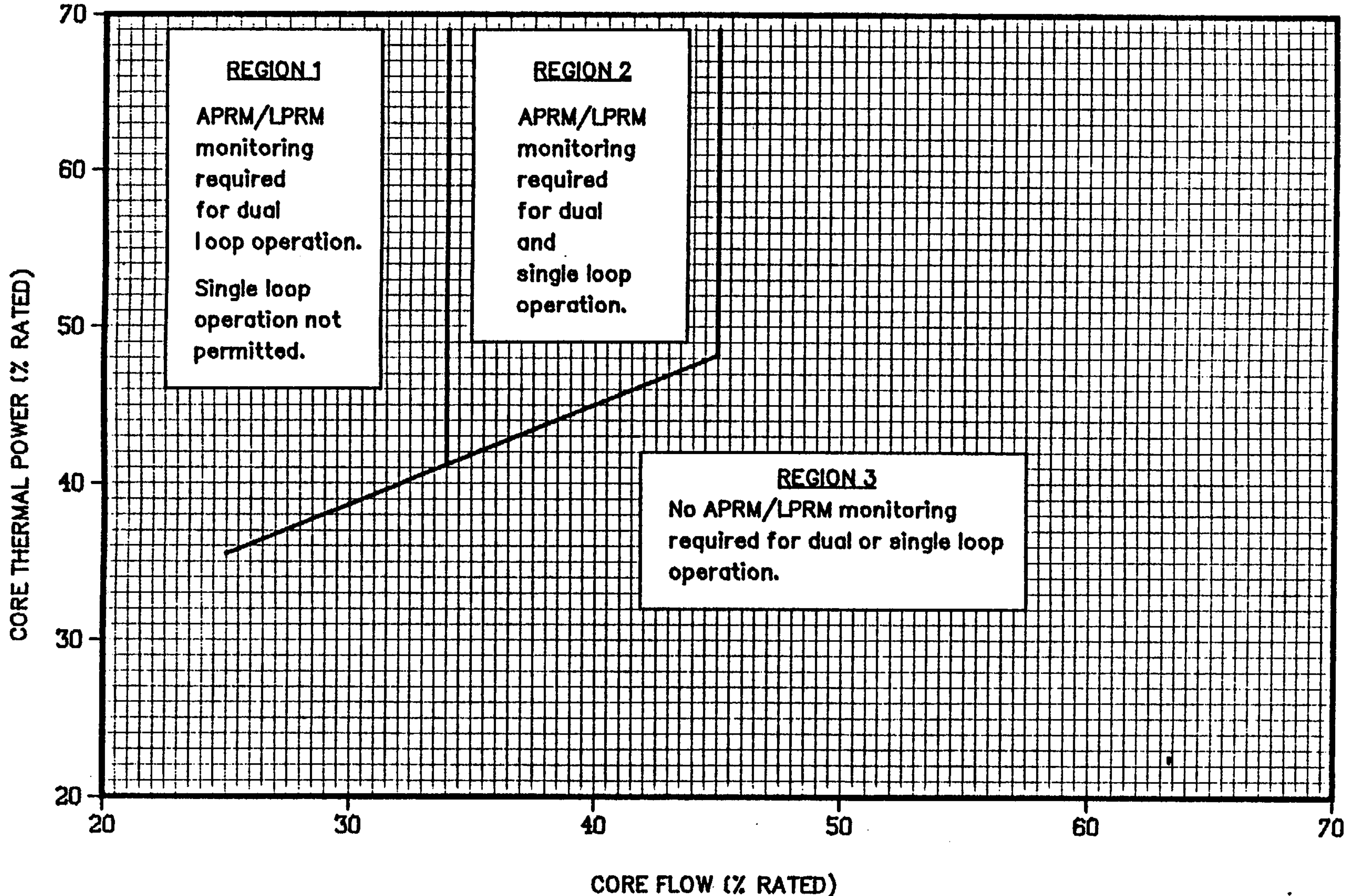
2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

J. Thermal Hydraulic Stability

Figure 3.6.4

THERMAL POWER AND CORE FLOW LIMITS FOR

APRM/LPRM MONITORING



3.6 & 4.6 (continued)

The following factors form the basis for the surveillance requirements:

A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Specifications 4.6.F.1 and 2.

Agreement of indicated core flow with established power-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the twenty individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus, the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle-riser system failure.

3.6 & 4.6 (continued)

G. Single Loop Operation

Continuous operation with one recirculation loop was justified in "Vermont Yankee Nuclear Power Station Single Loop Operation", NEDO-30060, February 1983, with the adjustments specified in Technical Specification 3.6.G.1.a.

APRM and/or LPRM oscillations in excess of those specified in Section 3.6.G.1.b could be an indication that a condition of thermal hydraulic/neutronic instability exists and that appropriate remedial action should be taken. By restricting core flow to greater than or equal to 34% of rated, which corresponds to the core flow at the 80% rod line with 2 recirculation pumps running at minimum speed, the region of the power/flow map where these oscillations are most likely to occur is avoided (Region 1 of Figure 3.6.4). These specifications are based upon the guidance of GE SIL #380, Revision 1, dated February 10, 1984.

During single loop operation, the idle recirculation loop is isolated by electrically disarming the recirculation pump motor generator set drive motor, until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Under single loop operation, the flow control is placed in the manual mode to avoid control oscillations which may occur in the recirculation flow control system under these conditions.

3.6 & 4.6 (continued)

H. Recirculation System

The largest recirculation break area assumed in the ECCS evaluation was 4.14 square feet.

APRM and/or LPRM oscillations in excess of those specified in Section 3.6.H.1 could be an indication that a condition of thermal hydraulic instability exists and that appropriate remedial action should be taken. These specifications are based upon the guidance of GE SIL #380, Revision 1, dated February 10, 1984.

Specification 3.6.H.3 restricts reactor operation under natural circulation conditions in order to avoid potential thermal hydraulic/neutronic instabilities.

3.6.I & 4.6.I SHOCK SUPPRESSORS (SNUBBERS)

All snubbers are required operable to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are (1) of a specific make or model, (2) of the same design, and (3) similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration. These characteristics of the snubber installation shall be evaluated to determine if further functional testing of similar snubber installations is warranted.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested once each operating cycle. Observed failures of these sample snubbers shall require functional testing of additional units.

3.6.J THERMAL HYDRAULIC STABILITY

Not allowing startup of an idle recirculation pump from natural circulation conditions prevents the reactivity insertion transient that would occur.

LIMITING CONDITION FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting values shown in Tables 3.11-1A through G. For single recirculation loop operation, the limiting values shall be the values from Tables 3.11-1B through E and Table 3.11-1G listed under the heading "Single Loop Operation." These values are obtained by multiplying the values for two loop operation by 0.83. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within

SURVEILLANCE REQUIREMENT

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at >25% rated thermal power.

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

C. Minimum Critical Power Ratio (MCPR)

1. During steady-state power operation, the MCPR Operating Limit shall be equal or greater than the values shown on Table 3.11-2. For single recirculation loop operation, the MCPR Limits at rated flow are increased by 0.01. For core flows other than rated, the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is given by Figure 3.11-2. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor power shall be brought to shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

Bases:

3.11 Fuel Rods

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

Refer to Section 5.5.2 of NEDE-24011P, Amendment 3, dated March 1978.

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

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1.

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.

Bases:

3.11C 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 presented in Appendix A of the current cycle's Core Performance Analysis Report. The 0.01 increase in MCPR operating limits for single loop operation accounts for increased core flow measurement and TIP reading uncertainties, as discussed in NEDO-30060, "Vermont Yankee Nuclear Power Station Single Loop Operation", February, 1983.

Table 3.11-1B

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee Fuel Type: 8D219

Average Planar Exposure (MWD/t)	MAPLHGR (kW/ft) Two Loop Operation	*Single Loop Operation	PCT (^{o}F)	Oxidation Fraction
200.0	11.4	9.5	2053.	0.021
1,000.0	11.5	9.5	2061.	0.021
5,000.0	11.9	9.9	2117.	0.023
10,000.0	12.1	10.0	2164.	0.026
15,000.0	12.3	10.2	2192.	0.029
20,000.0	12.1	10.0	2189.	0.029
25,000.0	11.3	9.4	2077.	0.020
30,000.0	10.2	8.5	1933.	0.012
35,000.0	9.6	8.0	1704.	0.004

Source: NEDO-21697, August 1977 (revised)

*MAPLHGR for single loop operation is obtained by multiplying MAPLHGR for two loop operation by 0.83.

Table 3.11-1C

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee Fuel Type: 8D274L

Average Planar Exposure (MWD/t)	MAPLHGR (kW/ft) Two Loop Operation	*Single Loop Operation	PCT (^{o}F)	Oxidation Fraction
200.0	11.2	9.3	2060.	0.019
1,000.0	11.3	9.4	2064.	0.019
5,000.0	11.9	9.9	2133.	0.024
10,000.0	12.1	10.0	2129.	0.023
15,000.0	12.2	10.1	2159.	0.025
20,000.0	12.1	10.0	2167.	0.026
25,000.0	11.6	9.6	2118.	0.023
30,000.0	10.9	9.0	2028.	0.017
35,000.0	9.9	8.2	1896.	0.010
40,000.0	9.3	7.7	1812	0.007

Source: NEDO-21697, August 1977 (revised)

*MAPLHGR for single loop operation is obtained by multiplying MAPLHGR for two loop operation by 0.83.

Table 3.11-1D

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee   Fuel Type: 8D274H

Average Planar Exposure (MWD/t)	MAPLHGR (kW/ft) Two Loop Operation	*Single Loop Operation	PCT (°F)	Oxidation Fraction
200.0	11.1	9.2	2052.	0.019
1,000.0	11.2	9.3	2050.	0.018
5,000.0	11.8	9.8	2113.	0.022
10,000.0	12.1	10.0	2131.	0.023
15,000.0	12.2	10.1	2161.	0.026
20,000.0	12.0	10.0	2164.	0.026
25,000.0	11.5	9.5	2112.	0.022
30,000.0	10.9	9.0	2029.	0.017
35,000.0	10.0	8.3	1900.	0.011
40,000.0	9.3	7.7	1815.	0.008

Source: NEDO-21697, August 1977 (revised)

*MAPLHGR for single loop operation is obtained by multiplying MAPLHGR for two loop operation by 0.83.

Table 3.11-1E

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee　　Fuel Type: 8D274 (High Gd)

Average Planar Exposure (MWD/t)	MAPLHGR (kW/ft) Two Loop Operation	MAPLHGR (kW/ft) *Single Loop Operation	PCT (°F)	Oxidation Fraction
200.0	11.1	9.2	2053.	0.019
1,000.0	11.1	9.2	2044.	0.018
5,000.0	11.6	9.6	2092.	0.021
10,000.0	12.1	10.0	2141.	0.024
15,000.0	12.2	10.1	2165.	0.026
20,000.0	12.1	10.0	2170.	0.027
25,000.0	11.6	9.6	2119.	0.023
30,000.0	10.6	8.8	1993.	0.015
35,000.0	10.0	8.3	1751.	0.005
40,000.0	9.4	7.8	1671.	0.004

Source: NEDO-21697, August 1977 (revised)

*MAPLHGR for single loop operation is obtained by multiplying MAPLHGR for two loop operation by 0.83.

Table 3.11-1G

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee | Fuel Type: 8DPB289 & P8DPB289

Average Planar Exposure (MWD/t)	MAPLHGR (kW/ft) Two Loop Operation	*Single Loop Operation	PCT (°F)	Oxidation Fraction
200.0	11.2	9.3	2126.	0.027
1,000.0	11.2	9.3	2119.	0.026
5,000.0	11.8	9.8	2178.	0.030
10,000.0	12.0	10.0	2185.	0.030
15,000.0	12.1	10.0	2200.	0.032
20,000.0	11.8	9.8	2187.	0.031
25,000.0	11.3	9.4	2120.	0.025
30,000.0	11.1	9.2	2095.	0.023
35,000.0	10.4	8.6	1862.	0.008
40,000.0	9.8	8.1	1784.	0.006

Source: NEDO-21697, August 1977 (revised)

*MAPLHGR for single loop operation is obtained by multiplying MAPLHGR for two loop operation by 0.83.

Table 3.11-2
MCPR Operating Limits (3)

Value of "N" in RBM Equation (1)	Average Control Rod Scram Time	Cycle Exposure Range	MCPR Operating Limit for Fuel Type (2) 8x8	8x8R	P8x8R
42%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.29	1.29	1.29
		EOC-2 GWD/T to EOC-1 GWD/T	1.29	1.29	1.29
		EOC-1 GWD/T to EOC	1.30	1.30	1.30
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.29	1.29	1.29
		EOC-2 GWD/T to EOC-1 GWD/T	1.33	1.31	1.31
		EOC-1 GWD/T to EOC	1.36	1.35	1.35
41%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.25	1.25	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.26	1.25	1.25
		EOC-1 GWD/T to EOC	1.30	1.30	1.30
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.25	1.25	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.33	1.31	1.31
		EOC-1 GWD/T to EOC	1.36	1.35	1.35
≤40%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.25	1.25	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.26	1.25	1.25
		EOC-1 GWD/T to EOC	1.30	1.30	1.30
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.25	1.25	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.33	1.31	1.31
		EOC-1 GWD/T to EOC	1.36	1.35	1.35

(1) The Rod Block Monitor (RBM) trip setpoints are determined by the equation shown in Table 3.2.5 of the Technical Specifications.

(2) The current analyses for MCPR Operating Limits do not include 7x7 fuel. On this basis, further evaluation of MCPR Operating Limits is required before 7x7 fuel can be used in reactor power operation.

(3) MCPR Operating Limits are increased by 0.01 for single loop operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 94 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 from W. Murphy, Vermont Yankee Nuclear Power Corporation, to H. Denton, Nuclear Regulatory Commission (Reference 1), Technical Specification (TS) changes were proposed for Vermont Yankee Nuclear Power Station (VY) to permit extended operation with only one recirculation pump in operation and to address thermal-hydraulic stability monitoring during this Single Loop Operation (SLO) as well as in Two Loop Operation (TLO). Following discussions with the staff, there was a second submittal by VY (Reference 2) revising some details of the TS relating to SLO limits and Thermal-Hydraulic Stability (THS). Also submitted was a letter (Reference 3) enclosing a report (Reference 4) by General Electric (GE) providing an analysis of SLO for VY.

VY current TS permit SLO only for 24 hours and there are no provisions for potential alteration of safety and operating limits because of SLO conditions, and no provisions for THS monitoring, if needed, either for SLO or TLO. The proposed changes would permit SLO for unlimited time frames and without specific additional power limits, and provide for needed changes to operating limits and require THS monitoring. A primary resistance to proposals by VY and others to extend SLO has been NRC concerns relating to THS analyses and monitoring. These concerns have been largely resolved in the past year, and a number of plants have proposed TS changes to permit SLO and these have been reviewed and accepted by the staff. Principle examples of these changes and acceptances may be found in References 5 and 6, the Safety Evaluation Reports for Duane Arnold, a lead plant for the changes, and for Susquehanna, the most recent example of evolving TS.

The staff approved resolution of the THS problem is based primarily on the approval of the GE surveillance mode of "detect and suppress" expressed in the GE Service Information Letter (SIL) 380 (Reference 7). This mode of operation may be used for both TLO (instead of providing approved THS analysis for each reload cycle) and for SLO. This mode of operation is now required by the staff for SLO without specific regard to THS analysis. The NRC has recently published two generic letters (References 8 and 9) relating to THS and SLO. These generic letters present staff positions in this area including acceptance of surveillance modes recommended in SIL-380 and provide a basis for review of TS changes.

The VY proposal is similar to the previously approved requests for SLO and TLO "detect and suppress" TS changes and is in accord with staff positions in the generic letters. VY will use surveillance for TLO as well as SLO and will not provide THS analyses for each cycle.

2.0 EVALUATION

There are two principal areas related to SLO which require TS changes. They are (1) changes to some safety and operating limits and instrument setpoints resulting from altered reactor conditions and reanalysis of relevant transient and accident events, and (2) operating limits related to the power-flow map location and monitoring of THS and the suppression of possible oscillations. The VY original submittal (Reference 1) presented TS changes and supporting information in both areas. The supporting information for changes to limits and setpoints included the report by GE (Reference 4) of the analyses of relevant transients and accidents for SLO for VY and the derivation of appropriate limits and operating conditions. The changes to provide monitoring and suppression of oscillations were based on SIL-380 and generally patterned after the Duane Arnold TS. After the initial submittal there were several discussions with the staff, largely related to required time frames for limit resetting actions and monitoring related activities and these resulted in a supplemental submittal (Reference 2) altering some of the time related values as well as some flow restrictions and baseline requirements.

The GE analyses of SLO (Reference 4) provide the following changes to Safety Limit (SL) and Operating Limit (OL) Minimum Critical Power Ratio (MCPR) and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits, and to Average Power Range Monitor (APRM) and Rod Block Monitor (RBM) setpoints. These analyses and changes are similar in content and magnitude to those approved in previous reviews.

(1) SL MCPR and OL MCPR: The SL MCPR is increased by 0.01 to account for increased uncertainties in core flow and power distribution measurements. The OL MCPR is increased for VY by the same amount. Limiting transients were reexamined to assure that this is a sufficient increase throughout the achievable power-flow range. No additional increase in OL MCPR is necessary since transient events initiated from TLO remain limiting. These changes are acceptable.

(2) APRM scram and rod block and RBM flow biased limit functions: These equations have an added term to account for the difference between single and two loop drive flow for the same core flow. This adjustment accounts for the difference between actual and indicated flow and preserves the original relation between limits and effective drive flow. These changes are acceptable.

(3) MAPLHGR: GE has performed a series of LOCA analyses for VY to provide a MAPLHGR multiplier for this event. The principle change in the analysis is the assumption of an earlier occurring boiling transition. The multiplier for VY for all GE fuel was determined to be 0.83. This is acceptable.

Other relevant transients and accidents were examined and evaluated and found to require no additional changes to the TS. This included an analysis of the SLO recirculation pump seizure accident for which it was found that the minimum MCPR remained greater than the SL MCPR by 0.14 (Reference 10). This analysis is acceptable.

In addition to these analyses GE recommends that the flow control should be in manual. This condition is in the TS.

To handle possible problems of THS in TLO or SLO VY has provided new TS providing operating restrictions and/or monitoring of APRM and selected Local Power Range Monitor (LPRM) noise levels when in certain regions of the power-flow map. These requirements are in accord with the recommendation of SIL-380 and, as worked out in discussions with the staff, are similar in content to THS restrictions and monitoring approved in the past year (e.g., Reference 5 and 6).

A principle feature of the restrictions is the region in the power-flow map above the 80 percent power-flow line and between 34 and 45 percent flow in which APRM and LPRM monitoring is required (34 percent flow corresponds to two pump minimum flow for VY). Below 34 percent flow (and above the 80 percent line) SLO is not permitted. Baseline noise levels are required and in the monitoring region action must be taken to suppress oscillations if the noise levels are greater than 3 times the baseline levels. Appropriate time intervals are set up in which to carry out these operations. These action and surveillance TS for THS are acceptable.

Since VY will monitor THS in TLO as well as SLO they are not required to submit a stability analysis for this or future reloads.

The initial submittal had a core flow (maximum) limit of 45 percent for SLO to address flow-induced core plate delta p noise. In response to staff discussions this was altered to require only an administrative limit requiring procedures such that SLO above 50 percent core flow would involve monitoring core plate delta p noise. This is an acceptable alternative.

Although the subject is not discussed in the VY submittal, it may be noted that some general questions have arisen about the adequacy of SLO surveillance methods which have been used in some plants to monitor jet pump operability in accordance with NUREG/CR-3052 to close out problems presented in IE Bulletin 80-07, "BWR Jet Pump Assembly Failure." Since VY has replaced all 20 of the hold down beams in question with an acceptable improved design, the question regarding SLO jet pump surveillance adequacy is not applicable.

3.0 TECHNICAL SPECIFICATION CHANGES

To accomplish the required changes for SLO limits and THS monitoring the following TS have been changed. (There are also minor administrative changes). Most of the specifications were altered in the second submittal (Reference 2) to change action and surveillance times and other THS factors previously discussed.

(1) 1.1 and Basis 1.1: The SL MCPR has been increased by 0.01 to 1.08 for SLO because of an increased noise uncertainty contribution. This is acceptable.

(2) 2.1.A, 2.1.B, Figure 2.1-1 and Basis 2.1: The equation for the APRM Flux Scram setting and APRM Rod Block trip setting have an added delta W term to account for the difference between TLO and SLO drive flow at the same core flow. This has been previously discussed and is acceptable. There are also some administrative page changes to accommodate the technical changes.

(3) Tables 3.1.1 and 3.2.5 and accompanying footnotes and Basis 3.2: These changes address the setpoint changes required because of flow changes for SLO discussed for TS 2.1 above. These tables provide the settings for the Reactor Protection System and Rod Block instrumentation. The changes add the same delta W term as in TS 2.1. They are acceptable.

(4) 3/4.6.G, H and J, Figure 3.6.4 and Bases 3/4.6.G, H and J: These changes and additions provide for and describe the TLO and SLO surveillance regions in the power-flow map, the nature of the APRM-LPRM surveillance, baseline and operation information required, and the levels and time frames for action in providing this surveillance and acting on it. Manual flow control is required for SLO. As previously discussed these requirements provide a satisfactory representation of the recommendation of SIL-380 and the staff generic position and previous approvals, and are acceptable. There are also some administrative page changes and a change relating to previously approved Amendment 92.

(5) 3.11.A and C, Bases 3.11.A and C and Tables 3.11-1B through 1E, 1G and 2: These changes provide for the decrease in the MAPLHGR limit

by the factor 0.83 for SLO, and also for an increase in OL MCPR by 0.01. The changes are acceptable.

We have reviewed the reports submitted by VY proposing TS changes relating to SLO and THS. Based on this review we conclude that appropriate documentation was submitted and that the proposed changes satisfy staff positions and requirements in these areas. SLO operation and THS monitoring in the manner thus described is acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

This 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. 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: H. Richings

Dated: August 8, 1986

REFERENCES

1. Letter from W. Murphy, Vermont Yankee Nuclear Power Corporation (VYNPC), to H. Denton, NRC, dated March 12, 1986, "Proposed Technical Specification Change for Single Loop Operation...."

2. Letter from R. Capstick, VYNPC, to V. Rooney, NRC, dated May 9, 1986 "Single Loop Operation and Thermal-Hydraulic Stability"

3. Letter from R. Capstick, VYNPC, to V. Rooney, NRC, dated March 27, 1986, "...-Supporting Document"

4. General Electric Report NEDO-30060, "Vermont Yankee Single Loop Operation," dated February 1983.

5. Letter from M. Thadani, NRC, to L. Liu, Iowa Electric Light and Power Company, dated May 28, 1985, Docket No. 50-331.

6. Letter from E. Adensam, NRC, to H. Keiser, Pennsylvania Power and Light Company, dated April 11, 1986, "Amendment Nos. 56 and 26 ..... Susquehanna Units 1 and 2."

7. General Electric Service Information Letter No.380, Revision 1, February 10, 1984.

8. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19-Thermal Hydraulic Stability," January 23, 1986.

9. Generic Letter No. 86-09, "Technical Resolution of Generic Issue No. B-59-(N-1) Loop Operation in BWRs and PWRs," March 31, 1986.

10. Letter from R. Capstick, VYNPC, to V. Rooney, NRC, dated June 9, 1986, ".... Response to Request for Information."