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September 12, 2000
BVY 00-81

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: **Vermont Yankee Nuclear Power Station**
License No. DPR-28 (Docket No. 50-271)
Technical Specification Proposed Change No. 226 - Update
Control Rod Block Instrumentation

Reference: (a) Letter, VYNPC to USNRC, "Technical Specification Proposed Change No. 226, Control Rod Block Instrumentation," BVY 99-160, dated December 21, 1999

By Reference (a), Vermont Yankee (VY) requested an amendment to Facility Operating License, DPR-28, and its associated Technical Specifications (TS). The Proposed Change involves control rod block instrumentation requirements.

Following submittal of the subject Proposed Change, additional amendments to the Technical Specifications (i.e., Amendments 186, 187, and 193) have been approved by the Nuclear Regulatory Commission which impact nearly all of the pages covered by this Proposed Change. Because of this, VY is hereby providing replacement marked-up pages of the current TS, as well as revised re-typed pages, reflecting the current TS and Bases. These updated page changes fall within the scope of the original application, are administrative in nature with respect to the original Proposed Change, and are only being made to update and conform to the intervening TS amendments. The bases for the Proposed Change, safety considerations and determination of no significant safety hazard consideration are unchanged from those discussed in Reference (a).

Attachment 1 to this letter contains a discussion of differences between the page changes as proposed by Reference (a) and as finally made, considering the intervening license amendments. These updated changes do not alter the intent, meaning, or bases of the originally proposed changes. Attachment 2 provides the mark-up of the current Technical Specification and Bases pages. Attachment 3 is the retyped Technical Specification and Bases pages.

If you have any questions on this transmittal, please contact Mr. Thomas B. Silko at (802) 258-4146.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Gautam Sen

Gautam Sen
Licensing Manager

A001

Attachments

cc: USNRC Region 1 Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

Docket No. 50-271
BVY 00-81

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226 - Update

Control Rod Block Instrumentation

Description of Conforming Page Changes

INTRODUCTION

Subsequent to the original submittal of Proposed Change 226 (PC-226) on December 21, 1999 (VY letter BVY 99-160), the Nuclear Regulatory Commission has approved additional amendments to the VY operating license. Amendments 186, 187, and 193, in particular, have altered the content or format of nearly all of the pages covered by PC-226. Because of this, VY is providing replacement marked-up pages of the current Technical Specifications (TS), as well as revised, re-typed pages. These changes do not alter the original scope of PC-226 and are administrative in nature with respect to the original application. The changes are only being made to conform to the intervening TS amendments. The bases for the changes, safety considerations and determination of no significant safety hazard consideration are unchanged from those discussed in the original request for amendment, which is still pending.

PC-226 involves changes to TS and Bases pages 51, 52, 69, 74, 77, 78, and 90. A discussion of differences from the changes as originally proposed in VY letter BVY 99-160 follows:

DESCRIPTION OF CHANGES

Pages 51 and 52 (Table 3.2.5)

At the time of submittal of PC-226, TS Table 3.2.5 contained a Note 6, applicable to the Rod Block Monitor trip function. Note 6 was subsequently deleted through Amendment 186.

Amendment 186 added a "Note 10" under the column heading, "Minimum Number of Operable Instrument Channels per Trip System" in three places. In two of those places, the corresponding trip functions are being relocated from TS to the Technical Requirements Manual (TRM) under PC-226, and the subject note will be duplicated in the TRM. In the third instance, "Note 10" was added to be applicable to the Rod Block Monitor Upscale and Downscale trip functions. Note 10 will be retained in the TS as applicable to these functions and will also be applicable to the Rod Block Monitor Inop trip function being added by PC-226.

Amendment 186 also added a new "Note 11." Note 11 applies to the Scram Discharge Volume trip function, which is being relocated to the VY TRM under PC-226. Consequently, Note 11 will also be relocated and will no longer have application in the TS.

PC-226 had originally proposed adding two new Notes to Table 3.2.5 that were previously designated as Notes 10 and 11. Because of the addition of unrelated Notes 10 and 11 by Amendment 186, the PC-226-added Notes will be re-designated as Notes 12 and 13, respectively.

Amendment 187 added the qualifying statement, "with a maximum as defined in the COLR" under the Trip Setting for the rod block monitor upscale trip function, such that the trip setting is now expressed in TS Table 3.2.5 as "$\leq$ 0.66 (W-ΔW) +N with a maximum as defined in the COLR (Note 5)."

Pages 69 and 74

Amendment 186 replaced Note 1 in Table 4.2.5 with a specified test frequency of "every three months." This applied to the Average Power Range Monitor and Rod Block Monitor trip functions. The Average Power Range Monitor trip function will be relocated to the TRM under PC-226, along with the revised surveillance frequencies.

Page 77

Amendment 187 changed page 77 by incorporating some text rollover from the previous page and by the revision of two paragraphs. PC-226 affects the same two paragraphs revised by Amendment 187. Some of the changes made by Amendment 187 are no longer applicable because of the relocation of requirements to the TRM (i.e., APRM, IRM, SRM discussion). The revised page 77 being submitted herewith does not contain all of the text that was added by Amendment 187, but does incorporate the changes originally proposed as part of PC-226.

Page 78

Amendments 187 and 193 both revised page 78. The original PC-226 change has been incorporated into this page.

Page 90

Page 90 has not been changed since the original submittal of PC-226.

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Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226 - Update

Control Rod Block Instrumentation

Mark-up of the Current Technical Specifications

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

	Minimum Number of Operable Instrument Channels per Trip ~~System~~ Function	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		≤5 x 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~~ 2	a. Upscale (Flow Bias) (Note 7)			X	≤0.66(W-ΔW)+N with a maximum as defined in the COLR (Note 5)
	~~1~~ 2	b. Downscale (Note 7)			X	≥2/125 Full Scale
	~~1~~ 2	c. Inop (Note 7)			X	N/A
(Notes 10,11)	1 (per volume)	Scram Discharge Volume (LT-3-231A/G (S1))	X	X	X	≤12 Gallons
(Note 8)	~~1~~ 2	Trip System Logic	X	X	X	
(Note 13)	2	Reactor Mode Switch - Shutdown Position (Note 12)				N/A

Amendment No. ~~12~~, ~~25~~, ~~64~~, ~~66~~, ~~73~~, ~~76~~, ~~90~~, ~~94~~, ~~131~~, ~~164~~, ~~186~~, ~~187~~

Deleted <4 times>

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 ≥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 the Core Operating Limits Report 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. ΔW = 0 for two recirculation loop operation.

6. Not used.

7. The trip may be bypassed when the reactor power is ≤30% of ~~rated~~ Rated Thermal Power. 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 (as described in the Bases for Specification 3.3.B.6)

   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.

10. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required action notes may be delayed for up to 6 hours provided the associated Trip Function maintains Control Rod Block initiation capability.

Deleted.

11. ~~A. With the number of operable channels one less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within 12 hours.~~

    ~~B. With the number of operable channels two less than required by the minimum operable channels per trip function requirement, place the Trip System in the tripped condition within 1 hour.~~

INSERT #-NOTES 12 & 13 <originally submitted as Notes 10 & 11>

INSERT #1 (new Notes 12 and 13 to Table 3.2.5)

12. Required to be operable when the reactor mode switch is in the shutdown position.
13. With one or more Reactor Mode Switch – Shutdown Position channels inoperable, immediately suspend control rod withdrawal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

TABLE 4.2.5

MINIMUM TEST AND CALIBRATION FREQUENCIES

CONTROL ROD BLOCK INSTRUMENTATION

Trip Function	Functional Test	Calibration
Startup Range Monitor		
a. Upscale	Notes 4 and 6	Note 6
b. Detector Not Fully Inserted	Note 6	NA
Intermediate Range Monitor		
a. Upscale	Notes 4 and 6	Note 6
b. Downscale	Notes 4 and 6	Note 6
c. Detector Not Fully Inserted	Note 6	NA
Average Power Range Monitor		
a. Upscale (Flow Bias)	Every Three Months (Note 4)	Every Three Months
b. Downscale	Every Three Months (Note 4)	Every Three Months
Rod Block Monitor		
a. Upscale (Flow Bias)	Every Three Months (Note 4)	Every Three Months (Note 13)
b. Downscale	Every Three Months (Note 4)	Every Three Months
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)
High Water Level in Scram Discharge Volume	Every Three Months	Refueling Outage
c. Inop	Every Three Months	N/A
Reactor Mode Switch - Shutdown Position	Once/Operating Cycle (Note 12)	N/A

TABLE 4.2 NOTES

1. Not used.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart.

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.

4. This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

5. Deleted.

Deleted.

6. ~~Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibration shall be performed prior to or during each startup or controlled shutdown with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when instruments are required to be operable.~~

7. This instrumentation is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every three months.

8. Functional tests and calibrations are not required when systems are not required to be operable.

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

<INSERT 2-Notes 12 & 13 — as originally submitted>

INSERT #2 (new Notes 12 and 13 to Table 4.2.5)

12. Trip system logic is not applicable to this function. Must be performed within 1 hour after the reactor mode switch is placed in Shutdown, if surveillance is not current (once/operating cycle).
13. Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).

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 Rod Block Monitor (RBM) 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 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

<INSERT #3>

INSERT #3 (Bases 3.2, Page 77)

During hot shutdown and cold shutdown, and during refueling when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch--Shutdown Position control rod withdrawal block, required to be operable with the mode switch in the shutdown position, ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis. Two channels are required to be operable to ensure that no single channel failure will preclude a rod block when required. There is no trip setting for this function since the channels are mechanically actuated based solely on reactor mode switch position. During refueling with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock provides the required control rod withdrawal blocks.

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-30066, 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 set forth in the Offsite Dose Calculation Manual. 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: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system has no scram functions. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the withdrawal ~~of a designated single control rod could result in one or more fuel rods with MCPR less than the fuel cladding integrity safety limit.~~ During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.

a violation of the MCPR safety limit or the 1% plastic strain limit. A limiting control rod pattern is a pattern which results in the core being on a thermal limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

Continuous

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Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226 - Update

Control Rod Block Instrumentation

Retyped Technical Specification Pages

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip Function		Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
		Rod Block Monitor (RBM A/B)				
(Notes 10,9)	2	a. Upscale (Flow Bias) (Note 7)			X	$\leq$0.66(W-ΔW)+N with a maximum as defined in the COLR (Note 5)
	2	b. Downscale (Note 7)			X	$\geq$2/125 Full Scale
	2	c. Inop (Note 7)			X	N/A
(Note 13)	2	Reactor Mode Switch - Shutdown Position (Note 12)				N/A
(Note 8)	2	Trip System Logic	X	X	X	

TABLE 3.2.5 NOTES

1. Deleted.

2. Deleted.

3. Deleted.

4. Deleted.

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 the Core Operating Limits Report 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. Not used.

7. The trip may be bypassed when the reactor power is $\leq$30% of Rated Thermal Power. 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 (as described in the Bases for Specification 3.3.B.6), 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.

10. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into associated Limiting Conditions for Operation and required action notes may be delayed for up to 6 hours provided the associated Trip Function maintains Control Rod Block initiation capability.

11. Deleted.

12. Required to be operable when the reactor mode switch is in the shutdown position.

13. With one or more Reactor Mode Switch - Shutdown Position channels inoperable, immediately suspend control rod withdrawal and immediately initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.

TABLE 4.2.5

MINIMUM TEST AND CALIBRATION FREQUENCIES

CONTROL ROD BLOCK INSTRUMENTATION

Trip Function	Functional Test	Calibration
Rod Block Monitor		
a. Upscale (Flow Bias)	Every Three Months (Note 4)	Every Three Months (Note 13)
b. Downscale	Every Three Months (Note 4)	Every Three Months
c. Inop	Every Three Months	N/A
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)
Reactor Mode Switch - Shutdown Position	Once/Operating Cycle (Note 12)	N/A

TABLE 4.2 NOTES

1. Not used.

2. During each refueling outage, simulated automatic actuation which opens all pilot valves shall be performed such that each trip system logic can be verified independent of its redundant counterpart.

3. Trip system logic calibration shall include only time delay relays and timers necessary for proper functioning of the trip system.

4. This instrumentation is excepted from functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

5. Deleted.

6. Deleted.

7. This instrumentation is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every three months.

8. Functional tests and calibrations are not required when systems are not required to be operable.

9. The thermocouples associated with safety/relief valves and safety valve position, that may be used for back-up position indication, shall be verified to be operable every operating cycle.

10. Separate functional tests are not required for this instrumentation. The calibration and integrated ECCS tests which are performed once per operating cycle will adequately demonstrate proper equipment operation.

11. Trip system logic functional tests will include verification of operation of all automatic initiation inhibit switches by monitoring relay contact movement. Verification that the manual inhibit switches prevent opening all relief valves will be accomplished in conjunction with Section 4.5.F.1.

12. Trip system logic is not applicable to this function. Must be performed within 1 hour after the reactor mode switch is placed in Shutdown, if surveillance is not current (once/operating cycle).

13. Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).

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 Rod Block Monitor (RBM) 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 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.

During hot shutdown and cold shutdown, and during refueling when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch—Shutdown Position control rod withdrawal block, required to be operable with the mode switch in the shutdown position, ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby

BASES: 3.2 (Cont'd)

preserving the assumptions of the safety analysis. Two channels are required to be operable to ensure that no single channel failure will preclude a rod block when required. There is no trip setting for this function since the channels are mechanically actuated based solely on reactor mode switch position. During refueling with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock provides the required control rod withdrawal blocks.

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 set forth in the Offsite Dose Calculation Manual. 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: 3.3 & 4.3 (Cont'd)

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, (the latest NRC-approved version will be listed in the COLR).

5. The Source Range Monitor (SRM) system has no scram functions. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the continuous withdrawal of a designated single control rod could result in a violation of the MCPR safety limit or the 1% plastic strain limit. A limiting control rod pattern is a pattern which results in the core being on a thermal limit, i.e., operating on a limiting valve for APLHGR, LHGR, or MCPR. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.