VERMONT YANKEE NUCLEAR POWER CORPORATION

185 Old Ferry Road, Brattleboro, VT 05301-7002
(802) 257-5271

December 21, 1999
BVY 99-160

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
<u>Control Rod Block Instrumentation</u>

Pursuant to 10CFR50.90, Vermont Yankee (VY) hereby proposes to amend its Facility Operating License, DPR-28, by incorporating the attached proposed change into the VY Technical Specifications. This proposed change revises the control rod block requirements consistent with the BWR/4 Standard Technical Specifications[1]. In so doing, some functions are proposed to be relocated to the Technical Requirements Manual, the requirements for the retained functions are clarified, and two functions are added to the Technical Specifications.

Attachment 1 to this letter contains supporting information and the safety assessment of the proposed change. Attachment 2 contains the determination of no significant hazards consideration. Attachment 3 provides the marked-up version of the current Technical Specification and Bases pages. Attachment 4 is the retyped Technical Specification and Bases pages.

VY has reviewed the proposed Technical Specification change in accordance with 10CFR50.92 and concludes that the proposed change does not involve a significant hazards consideration.

VY has also determined that the proposed change satisfies the criteria for a categorical exclusion in accordance with 10CFR51.22(c)(9) and does not require an environmental review. Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment needs to be prepared for this change.

Upon acceptance of this proposed change by the NRC, VY requests that a license amendment be issued for implementation within 60 days of the effective date of the amendment.

A001 1/1

1 NUREG 1433, Revision 1, Standard Technical Specifications General Electric Plants, BWR/4, dated April 7, 1995

PDR ADOCK 05000271

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

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Robert J. Wanczyk

Robert J. Wanczyk
Director of Safety and Regulatory Affairs

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Robert J. Wanczyk, who, being duly sworn, did state that he is Director of Safety and Regulatory Affairs of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing document in the name and on the behalf of Vermont Yankee Nuclear Power Corporation, and that the statements therein are true to the best of his knowledge and belief.

SALLY A. SANDSTRUM NOTARY WINDHAM COUNTY, VT

Sally A. Sandstrum

Sally A. Sandstrum, Notary Public
My Commission Expires February 10, 2003

Attachments

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

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226

Control Rod Block Instrumentation

Supporting Information and Safety Assessment of Proposed Change

INTRODUCTION

The purpose of this proposed change is to clarify the Technical Specification control rod block instrumentation requirements. Those requirements are found in Table 3.2.5, Control Rod Block Instrumentation. The associated Surveillance Requirements are found in Table 4.2.5, Minimum Test and Calibration Frequencies Control Rod Block Instrumentation. The proposed change is consistent with the BWR/4 Standard Technical Specifications.

SAFETY ASSESSMENT

Revising the control rod block instrumentation requirements and incorporating the associated changes consistent with the BWR/4 Standard Technical Specifications is not a safety concern as explained in the following Table 1. Table 1 provides details of each change, including the basis for the change and a safety assessment. The Change Numbers in the left-hand column correspond to the boxed (□) annotation numbers in Attachment 3, Marked-Up Version of the Current Technical Specifications.

Table 1

<table>
<tr><th>Change #</th><th>Current Technical Specification</th><th>Proposed Change</th></tr>
<tr><td>1</td><td>Table 3.2.5: Column heading states “Minimum Number of Operable Instrument Channels per Trip System.”</td><td>Column heading changed to “Minimum Number of Operable Channels per Trip Function.” Number of channels in the column for those functions retained in the Technical Specifications are adjusted accordingly.</td></tr>
<tr><td colspan="3">Basis / Safety Assessment: The control rod block logic arrangement does not correspond to the RPS trip system arrangement. Control rod block logic is best described as “one out of n;” i.e., any one trip from a rod block channel results in a rod block. Therefore, it is appropriate to define the total number of channels required to satisfy single failure criteria.

Since this is an administrative change that clarifies the specification, there is no impact on safety.</td></tr>
</table>

Change #	Current Technical Specification	Proposed Change
2	Table 3.2.5 and Table 4.2.5 contain Startup Range Monitor Upscale and Detector Not Fully Inserted rod block functions.	Relocates these functions with their respective operability requirements and associated surveillance requirements to the Technical Requirements Manual (TRM).

Basis / Safety Assessment: The Source Range Monitor (SRM) control rod blocks function to prevent a control rod withdrawal error during reactor startup utilizing SRM signals to create the rod block signal. SRM signals are used to monitor neutron flux during refueling, shutdown and startup conditions. No design basis accident (DBA) or transient analysis takes credit for rod block signals initiated by the SRMs.

Comparing the SRM control rod block functions to the screening criteria of 10CFR50.36:

1. The SRM control rod block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The SRM control rod block instrumentation does not monitor a process variable that is an initial condition of a DBA or transient analysis.
3. The SRM control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 137) of NEDO-31466[2], the loss of the SRM control rod block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. VY has reviewed this evaluation, considers it to be applicable to VY, and concurs with the assessment.

Based on #1-4 above, the SRM control rod block functions can be relocated to the TRM with no impact on safety.

[2] NEDO 31466, Technical Specification Screening Criteria Application and Risk Assessment, dated November 1987.

Change #	Current Technical Specification	Proposed Change
3	Table 3.2.5 and Table 4.2.5 contain Intermediate Range Monitor Upscale, Downscale, and Detector Not Fully Inserted rod block functions.	Relocates these functions with their respective operability requirements and associated surveillance requirements to the TRM.

Basis / Safety Assessment: The Intermediate Range Monitor (IRM) control rod blocks function to prevent a control rod withdrawal error during reactor startup utilizing IRM signals to create the rod block signal. IRMs are provided to monitor the neutron flux levels during refueling, shutdown, and startup conditions. No DBA or transient analysis takes credit for rod block signals initiated by the IRMs.

Comparing the IRM control rod block functions to the screening criteria of 10CFR50.36:

1. The IRM control rod block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The IRM control rod block instrumentation does not monitor a process variable that is an initial condition of a DBA or transient analysis.
3. The IRM control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 138) of NEDO-31466, the loss of the IRM Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. VY has reviewed this evaluation, considers it applicable to VY, and concurs with the assessment.

Based on #1-4 above, the IRM control rod block functions can be relocated to the TRM with no impact on safety.

Change #	Current Technical Specification	Proposed Change
4	Table 3.2.5 and Table 4.2.5 contain Average Power Range Monitor Upscale (Flow Bias) and Downscale rod block functions.	Relocates these functions with their respective operability requirements and associated surveillance requirements to the TRM.

Basis / Safety Assessment: The Average Power Range Monitor (APRM) control rod blocks function to prevent a control rod withdrawal error during power range operations utilizing LPRM signals to create the APRM rod block signal. APRMs provide information about the average core power and APRM rod blocks are not used to mitigate a DBA or transient.

Comparing the APRM control rod block functions to the screening criteria of 10CFR50.36:

1. The APRM control rod block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The APRM control rod block instrumentation does not monitor a process variable that is an initial condition of a DBA or transient analysis.
3. The APRM control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 135) of NEDO-31466, the loss of the APRM Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. VY has reviewed this evaluation, considers it applicable to VY, and concurs with the assessment.

Based on #1-4 above, the APRM control rod block functions can be relocated to the TRM with no impact on safety.

Change #	Current Technical Specification	Proposed Change
5	Table 3.2.5 and Table 4.2.5 contain the Scram Discharge Volume rod block function.	Relocates this function with its operability requirements and associated surveillance requirements to the TRM.

Basis / Safety Assessment: The Scram Discharge Volume (SDV) control rod block functions to prevent control rod withdrawals during power range operations, utilizing SDV signals to create the rod block signal if water is accumulating in the SDV. The purpose of measuring the SDV water level is to ensure that there is sufficient volume to contain the water discharged by the control rod drives during a scram, thus ensuring that the control rods will be able to insert fully. This rod block signal provides an indication to the operator that water is accumulating in the SDV and prevents further rod withdrawals. With continued water accumulation, a reactor protection system initiated scram signal will occur. Thus, the SDV water level rod block signal provides an opportunity for the operator to take action to avoid a subsequent scram. No DBA or transient takes credit for rod block signals initiated by the SDV instrumentation.

Comparing the SDV control rod block functions to the screening criteria of 10CFR50.36:

1. The SDV control rod block instrumentation is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA.
2. The SDV control rod block instrumentation does not monitor a process variable that is an initial condition of a DBA or transient analysis.
3. The SDV control rod block instrumentation is not a part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Item 139) of NEDO-31466, the loss of the SDV Control Rod Block function was found to be a non-significant risk contributor to core damage frequency and offsite releases. VY has reviewed this evaluation, considers it applicable to VY, and concurs with the assessment.

Based on #1-4 above, the SDV control rod block functions can be relocated to the TRM with no impact on safety.

Change #	Current Technical Specification	Proposed Change
6	There are no explicitly established requirements in the Technical Specifications for the Inop function of the Rod Block Monitor (RBM).	Operability and surveillance requirements are added to Tables 3.2.5 and 4.2.5, respectively. The new operability requirement for the Inop function of the RBM requires it to be OPERABLE when > 30% Rated Thermal Power. Also, a requirement for a quarterly CHANNEL FUNCTIONAL TEST of the function is added. Note 7 to Table 3.2.5 is also clarified by changing "...≤30% of rated." To "...≤30% of Rated Thermal Power." "Rated Thermal Power" is defined in the TS Definitions and is a more precise term. This is an administrative change for clarity. Note 9 to Table 3.2.5 is modified by adding the parenthetical expression, "(as described in the Bases for Specification 3.3.B.6)." In addition, the subject Bases for Specification 3.3.B.6 is being revised and clarified to define in greater detail what is meant by the term, "limiting control rod pattern." The change to note 9 and the Bases do not change any operating requirements, but are administrative in nature by providing additional clarifying detail.

Basis / Safety Assessment: Adding a requirement for the Inop function of the RBM is necessary to ensure safety analysis assumptions concerning the RBM are maintained.

This change serves to ensure the operability of the RBM and is more restrictive on plant operation; therefore, there is no negative impact on safety.

Change #	Current Technical Specification	Proposed Change
7	A rod block is currently not required when the reactor mode switch is in the shutdown position.	Requirement is added for the control rod block function of the reactor mode switch in shutdown. A Note is added establishing the requirement to immediately suspend control rod withdrawal and initiate action to insert all insertable control rods in core cells containing one or more fuel assemblies if one or more Reactor Mode Switch – Shutdown position channels is inoperable. Adds a once per operating cycle channel functional test requirement, which will require that this test be performed within 1 hour after the Reactor Mode Switch is placed in Shutdown, if the surveillance is not current. In addition, Note 12 to Table 4.2.5 clarifies that the trip system logic testing is not applicable to this function. A Bases discussion is provided for the added function.

Basis / Safety Assessment: This change is necessary, since during shutdown conditions, no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality.

This is a more restrictive change that serves to ensure shutdown rod block assumptions are valid; therefore, there is no negative impact on safety.

Change #	Current Technical Specification	Proposed Change
8	Lines exist between groups of functions within Table 3.2.5.	The lines between groups of functions are deleted.
Basis / Safety Assessment: The lines imply that the Trip System Logic function is associated only with the Scram Discharge Volume function. The Trip System Logic function should be associated with all of the control rod block functions on the Table; therefore, the lines are deleted. This change remedies a potential misinterpretation of the applicability of the Trip System Logic function; therefore, there is no negative impact on safety.		
Note: Current TS erroneously has a left-hand bracket which combines the scram discharge volume trip function with the trip system logic function. This bracket was drawn in error and incorporated as such as part of TS Amendment 94. Prior to Amendment 94, the bracket was properly drawn. With the relocation of the scram discharge volume trip function to the TRM (see change 5 above), this administrative error will be corrected.		
9	Table 3.2.5 contains Notes 1 through 4 and Table 4.2.5 contains Note 6 that are associated only with functions that are relocated to the TRM.	The Notes are relocated with their associated specifications to the TRM.
Basis / Safety Assessment: The Notes are only associated with relocated functions, and therefore must be relocated with the functions. This change is purely administrative and has no impact on safety.		

Change #	Current Technical Specification	Proposed Change
10	There is no explicit requirement to verify that the RBM Upscale (Flow Bias) function is not bypassed when >30% Rated Thermal Power.	Adds an explicit quarterly requirement to verify that the RBM Upscale (Flow Bias) function is not bypassed when >30% Rated Thermal Power. (This is added as Note 13 to Table 4.2.5.)
Basis / Safety Assessment: This Surveillance Requirement is necessary to ensure the RBM is operable when required. Plant procedures currently do perform this verification at the proposed frequency. This change ensures the RBM is operable when required and is more restrictive on plant operations; therefore, there is no negative impact on safety.		
11	The Bases provides a discussion for functions which are relocated to the TRM in this Proposed Change.	Relocates to the TRM the Bases discussions associated with the functions relocated to the TRM in this Proposed Change. Provides editorial corrections necessitated by relocating portions of the text.
Basis / Safety Assessment: The subject Bases text is only associated with relocated functions, and therefore must be relocated with the functions. This change is purely administrative and has no impact on safety.		

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226

Control Rod Block Instrumentation

Determination of No Significant Hazards Consideration

Determination of No Significant Hazards Consideration

Description of amendment request:

Each of the proposed changes can be categorized as one of the following:

1) a function relocated to the Technical Requirements Manual (TRM) that does not meet the criteria of 10CFR50.36 for retention in the Technical Specifications;
2) an imposition of more restrictive requirements driven by an effort for consistency with the BWR/4 Standard Technical Specifications;
3) a purely administrative change to add clarity or necessitated by relocating the associated Technical Specification.

NRC has previously found, in other applications, the acceptability of relocating the identified functions to the TRM. Relocation to the TRM of requirements that do not meet the criteria of 10CFR50.36 does not diminish the basic requirements. Since the TRM is under the purview of 10CFR50.59, any subsequent revisions to the requirements will be administratively controlled by those provisions.

Basis for no significant hazards determination:

Pursuant to 10CFR50.92, Vermont Yankee (VY) has reviewed the proposed change and concludes that the change does not involve a significant hazards consideration since the proposed change satisfies the criteria in 10CFR50.92(c).

1. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The relocated functions are not assumed as initial conditions for, nor are they credited in the mitigation of, any design basis accident or transient previously evaluated. Since reactor operation with these revised and relocated Specifications is fundamentally unchanged, no design or analytical acceptance criteria will be exceeded. As such, this change does not impact initiators of analyzed events nor assumed mitigation of design basis accident or transient events.

More stringent and purely administrative changes do not affect the initiation of any event, nor do they negatively impact the mitigation of any event. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

None of the proposed changes affects any parameters or conditions that could contribute to the initiation of an accident. No new accident modes are created since the manner in which the plant is operated is unchanged. No safety-related equipment or safety functions are altered as a result of these changes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The operation of Vermont Yankee Nuclear Power Station in accordance with the proposed amendment will not involve a significant reduction in a margin of safety.

There is no impact on equipment design or operation, and there are no changes being made to safety limits or safety system settings that would adversely affect plant safety as a result of the proposed changes. Since the changes have no effect on any safety analysis assumption or initial condition, the margins of safety in the safety analyses are maintained. In addition, neither administrative changes with no technical impact, nor the imposition of more stringent requirements have a negative impact on a margin of safety. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Summary No Significant Hazards Consideration:

On the basis of the above, VY has determined that operation of the facility in accordance with the proposed change does not involve a significant hazards consideration as defined in 10CFR50.92(c), in that it: (1) does not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) does not create the possibility of a new or different kind of accident from any accident previously evaluated; and (3) does not involve a significant reduction in a margin of safety.

In making this determination, Vermont Yankee has also reviewed the NRC examples of license amendments considered not likely to involve significant hazards considerations as provided in the final adoption of 10CFR50.92 published in the Federal Register, Volume 51, No. 44, dated March 6, 1986.

Docket No. 50-271
BVY 99-160

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226

Control Rod Block Instrumentation

Marked-up Version of the Current Technical Specifications

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip ~~System~~ Function 1	Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting	
	Startup Range Monitor					
2	a. Upscale (Note 2)(7-40(A-D))	X	X		$\leq 5 \times 10^5$ cps (Note 3)	2
2	b. Detector Not Fully Inserted (7-11(A-D)(LS-4))	X	X			
	Intermediate Range Monitor					
(Note 1) 2	a. Upscale (7-41(A-F))	X	X		$\leq$108/125 Full Scale	3
2	b. Downscale (Note 4) (7-41(A-F))	X	X		$\geq$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	$\leq$0.66(W-ΔW)+42% (Note 5)	4
2	b. Downscale			X	$\geq$2/125 Full Scale	
	Rod Block Monitor (Note 6) (RBM A/B)					
(Note 9) ~~1~~ 2 1	a. Upscale (Flow Bias) (Note 7)			X	$\leq$0.66(W-ΔW)+N (Note 5)	
~~1~~ 2	b. Downscale (Note 7)			X	$\geq$2/125 Full Scale	
8 2	c. Inop (Note 7)			X	N/A	6
(Note 8) 1 (per volume)	Scram Discharge Volume (LT-3-231A/C (S1))	X	X	X	$\leq$12 Gallons	5
1 ~~1~~ 2	Trip System Logic	X	X	X		
7 (Note 11) 2	Reactor Mode Switch - Shutdown Position (Note 10)				N/A	7

<Trip System Logic should be last function on the table>

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

Page 1 of 8

Insert "Deleted" for each

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 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. 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.

Rated Thermal Power 6

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 (as described in the Bases for Specification 3.3.B.6), and 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. Required to be operable when the reactor mode switch is in the shutdown position. 7

11. 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			2
a. Upscale	Notes 4 and 6	Note 6	
b. Detector Not Fully Inserted	Note 6	NA	
Intermediate Range Monitor			3
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			4
a. Upscale (Flow Bias)	Notes 1 and 4	Every Three Months	
b. Downscale	Notes 1 and 4	Every Three Months	
Rod Block Monitor			
a. Upscale (Flow Bias)	Notes 1 and 4	Every Three Months (Note 13)	<add> 10
b. Downscale	Notes 1 and 4	Every Three Months	
c. Inop	Every Three Months	N/A	6
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)	
High Water Level in Scram Discharge Volume	Every Three Months	Refueling Outage	5
Reactor Mode Switch – Shutdown Position	Once / Operating Cycle (Note 12)	N/A	<add> 7

Page 3 of 8

Amendment No. ~~106~~, ~~110~~, 136

TABLE 4.2 NOTES

1. Initially once per month; thereafter, a longer interval as determined by test results on this type of instrumentation.

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. 9

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.

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).

<add> 7

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

<add> 10

Amendment No. ~~63~~, ~~98~~, ~~105~~, ~~145~~, 164

BASES: 3.2 (Cont'd)

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.

Rod Block Monitor (RBM)

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.~~

11

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.

11

~~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 NEDE-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.~~

7

<Insert 1>

Insert 1, Bases 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.

11

BASES: 3.2 (Cont'd)

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

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

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

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

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

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

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.

continuous

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~~ 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. 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.

6

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 226

Control Rod Block Instrumentation

Retyped Technical Specification Pages

Listing of Affected Technical Specifications Pages

Replace the Vermont Yankee Nuclear Power Station Technical Specifications pages listed below with the revised pages. The revised pages contain vertical lines in the margin indicating the areas of change.

Remove	Insert
51	51
52	52
69	69
74	74
77	77
78	78
90	90

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 (Note 6) (RBM A/B)				
(Note 9)	2	a. Upscale (Flow Bias) (Note 7)			X	$\leq 0.66(W-\Delta W)+N$ (Note 5)
	2	b. Downscale (Note 7)			X	$\geq 2/125$ Full Scale
	2	c. Inop (Note 7)			X	N/A
(Note 11)	2	Reactor Mode Switch - Shutdown Position	(Note 10)			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. 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 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. Required to be operable when the reactor mode switch is in the shutdown position.

11. 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)	Notes 1 and 4	Every Three Months (Note 13)
b. Downscale	Notes 1 and 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. Initially once per month; thereafter, a longer interval as determined by test results on this type of instrumentation.

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)

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.

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 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)

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

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

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

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

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

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.