VERMONT YANKEE NUCLEAR POWER CORPORATION

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BVY 01-51

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. 247
Control Rod Block Instrumentation

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 instrumentation requirements contained in Technical Specifications 2.1.B, Figure 2.1.1, and Tables 3.2.5 and 4.2.5. In so doing, some control rod block trip functions are being relocated to the VY Technical Requirements Manual, the requirements for the retained trip functions are clarified, and two trip 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 pages. Attachment 4 is the retyped Technical Specification 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.

VY requests that a license amendment be issued no later than six months from the date of this letter for implementation within 90 days of its effective date.

A001

If you have any questions on this transmittal, please contact Mr. Jeffrey T. Meyer at (802) 258-4105.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Michael A. Balduzzi
Senior Vice President and Chief Nuclear Officer

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

Then personally appeared before me, Michael A. Balduzzi, who, being duly sworn, did state that he is Senior Vice President and Chief Nuclear Officer 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 Public
My Commission Expires February 10, 2003

SALLY A. SANDSTRUM NOTARY PUBLIC WINDHAM COUNTY, VERMONT

Attachments

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

Docket No. 50-271
BVY 01-51

Attachment 1

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 247

Control Rod Block Instrumentation

Supporting Information and Safety Assessment of Proposed Change

INTRODUCTION

Purpose

Proposed Change 247 revises the control rod block instrumentation requirements contained in Technical Specifications (TS) 2.1.B (APRM Rod Block Trip Setting), Figure 2.1-1 (APRM Flow Reference Scram and APRM Rod Block Settings), Table 3.2.5 (Control Rod Block Instrumentation), and Table 4.2.5 (Minimum Test and Calibration Frequencies, Control Rod Block Instrumentation). These changes are intended to (1) clarify requirements consistent with current practices; (2) relocate requirements to the Technical Requirements Manual (TRM) that are not essential to assure safe operation; and, (3) add new requirements to ensure operability of required functions. Conforming changes are also being made to other TS, as well as the TS Bases.

The relocation of requirements is based on considerations of 10CFR50.36 and NUREG-1433[1]. Clarifications are necessary to provide consistency with VY terminology and to preclude confusion and potential errors. The additional requirements are added for completeness, recognizing their importance to safety.

The following Table 1 provides a detailed discussion of the specific changes, including the bases for the changes and associated safety assessments.

BACKGROUND

VY Design Considerations

Control rods provide the primary means for control of reactivity changes at the Vermont Yankee Nuclear Power Station. Control rod block instrumentation is designed to assure that specified fuel design limits are not exceeded for postulated transients and accidents (see VY UFSAR Section 14.5.3). For example, when reactor power is >30% of rated thermal power, the rod block monitor (RBM) provides protection for control rod withdrawal error events, and during shutdown conditions the reactor mode switch in the shutdown position ensures that all control rods remain inserted to prevent inadvertent criticalities. In addition, other plant systems can also provide separate rod block signals to inhibit rod withdrawal.

The rod block circuitry is arranged as two, parallel circuits, but as one trip system. These circuits are energized when control rod movement is allowed. The circuitry is powered from a 120 VAC instrument bus and receives input trip signals from a number of trip channels.

The Reactor Mode Switch, when placed in the shutdown position, provides signals through two channels, each inputting into a separate rod block circuit. This function ensures that the reactor remains subcritical during shutdown operations.

The design of the VY control rod block logic is "one-out-of-n, taken once" where any single channel trip will result in a rod block. Unlike most safety-related plant trip system logics (such as reactor protection system), the control rod block trip functions do not correspond to separate trip systems. The two rod block circuits receive inputs from various systems, for example, neutron monitoring. Rather than referring to control rod block instrumentation as being arranged in separate "trip systems," it is more appropriate to describe the operability requirements for control rod block instrumentation in terms of "required channels,"

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

that input to the rod block logic circuitry, since the logic used to generate a control rod block is contained within a single trip system. Defining operability requirements in terms of "required channels" is consistent with BWR industry practices as reflected in Standard Technical Specifications.

Comparison to Standard Technical Specifications

Standard Technical Specifications (STS) contain requirements for control rod block instrumentation in Section 3.3.2.1. Additional requirements related to control rod interlocks during refueling and special operations are contained in TS Sections 3.9 and 3.10.

In addition to the control rod block instrumentation requirements being addressed by this Proposed Change, VY's current TS also contain requirements regarding control rod block functions in Sections 3/4.3.B.3 and 3/4.3.B.6. Section 3.12.A contains requirements for control rod blocks related to refueling interlocks. Furthermore, TS 1.0.O provides definitions of VY protective instrumentation that are applicable to control rod block instrumentation.

The changes proposed in this revision of VY TS are consistent with STS.

Updated Final Safety Analysis Report (UFSAR)

The following VY UFSAR sections provide additional background information:

- 3.4.5.3 – Control Rod Drive Hydraulic System (including Scram Discharge Volume)
- 7.5.4 – Source Range Monitor Subsystem
- 7.5.5 – Intermediate Range Monitor Subsystem
- 7.5.7 – Average Power Range Monitor Subsystem
- 7.5.8 – Rod Block Monitor Subsystem
- 7.7.4.3 – Rod Block Interlocks
- 14.5.3 – Describes postulated events resulting in a positive reactivity insertion

SAFETY ASSESSMENT

Table 1 (below) provides a detailed description 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." Attachment 4, "Retyped Technical Specification Pages," illustrates the proposed changes in final form.

Table 1

Change #	Current Technical Specification	Proposed Change
1	Current Technical Specification (CTS) 2.1.B, "APRM Rod Block Trip Setting," and Figure 2.1-1, "APRM Flow Reference Scram and APRM Rod Block Settings," specify APRM flow biased rod block settings when the reactor mode switch is in the RUN position. TS 3.1.B.a, 4.1.B, 3.6.G.1.a, and 6.6.C.4 refer to TS 2.1.B and/or APRM rod block trip setting.	Relocate CTS 2.1.B and the associated portion of Figure 2.1-1 to the VY Technical Requirements Manual (TRM). Make conforming changes to TS 3.1.B.a, 4.1.B, 3.6.G.1.a, and 6.6.C.4 by deleting references to TS 2.1.B and APRM rod block setting. TS 3.1.B.a is changed to: *The APRM System gains shall be adjusted by the ratio given in Technical Specification 2.1.A.1, or* TS 4.1.B is changed to: *Once within 12 hours after ≥25% Rated Thermal Power and once a day during operation at ≥25% Rated Thermal Power thereafter, the maximum fraction of limiting power density and fraction of rated power shall be determined and the APRM system gains shall be adjusted by the ratio given in Technical Specification 2.1.A.1.a.* TS 3.6.G.1.a is changed to: *The designated adjustments for APRM flux scram setting (Specification 2.1.A.1.a and Table 3.1.1), rod block monitor trip setting (Table 3.2.5), MCPR fuel cladding integrity safety limit (Specification 1.1.A), and MCPR operating limits and MAPLHGR limits, provided in the Core Operating Limits Report, are initiated within 8 hours. During the next 12 hours, either these adjustments must be completed or the reactor brought to Hot Shutdown.* TS 6.6.C.4 is changed to: *The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a and 3.11.B, and*

Table 1
(continued)

Change #	Basis / Safety Assessment
1	The Average Power Range Monitor (APRM) control rod blocks function to prevent a control rod withdrawal error during reactor power operations utilizing local power range monitoring signals to create the APRM rod block signal. APRMs provide information about the average core power. No design basis accidents (DBA) or transient analyses take credit for rod block signals initiated by APRM instrumentation. Comparing these control rod block functions to the screening criteria of 10CFR50.36 for inclusion in TS: 1. The APRM control rod block instrumentation is not used for, nor is 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. 3. The APRM control rod block instrumentation is not 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[2], 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 also considers loss of the APRM control rod block function to be a non-significant risk contributor. This change is acceptable based on these four criteria, and the APRM control rod block function can be relocated to the TRM with no negative impact on plant safety. In addition, this is consistent with NUREG-1433. Conforming changes to other TS are administrative in nature and make no further changes to technical requirements.

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

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
2	The left-hand column of Table 3.2.5 has a heading, "Minimum Number of Operable Instrument Channels per Trip System." Numbers of operable channels are specified in this column for the corresponding Trip Functions listed in the second column of Table 3.2.5. In the second column of Table 3.2.5, under the heading "Trip Function," is a trip function identified as "Trip System Logic." Note 8 to Table 3.2.5 specifies protective action to be taken in the event that less than the minimum number of channels are operable.	The left-hand column heading of Table 3.2.5 is changed to "Required Channels." Numbers of channels in the column for RBM Upscale and Downscale functions are adjusted (i.e., increased by one) accordingly. This change results in the requirement for a minimum of two operable instrument channels for each of the RBM trip functions. "Trip System Logic" under the column heading "Trip Function" in TS Table 3.2.5 is changed to "Trip System." Note 8 to Table 3.2.5 is modified to clarify terminology consistent with the heading change. Note 8 is changed to: *With the number of operable channels less than the required number, place the inoperable channel in the tripped condition within one hour.*

Table 1
(continued)

Change #	Basis / Safety Assessment
2	The control rod block circuitry design does not correspond to multiple trip systems (as, for example, the reactor protection system). VY's control rod block logic is "one-out-of-n, taken once." That is, any one trip from a rod block channel results in a rod block. Therefore, the operability requirements for control rod block instrumentation are better described in terms of "required channels" since the logic scheme used to generate control rod blocks is actually a single trip system containing two control rod block circuits. See above "Background" for additional discussion. This revision is also consistent with NUREG-1433, Limiting Condition for Operation 3.3.2.1 and Table 3.3.2.1-1 in that requirements for control rod block instrumentation are specified in terms of "required channels." With the above changed designation, it is necessary to specify the required number of operable instrument channels as two, instead of one, for the RBM trip functions. (Note, other rod block trip functions are being removed from TS—see Change #3 and Change #5 below. And, other control rod block trip functions are being added—see Change #4 and Change #6 below.) Although a single operable channel is adequate to satisfy the safety function of this instrumentation, requiring two channels operable provides redundancy and ensures that no single instrument failure can preclude a rod block signal for the identified functions. The term "Trip System Logic" under the heading of "Trip Function" in Table 3.2.5 is changed to "Trip System" to better define system operability requirements. Since there is only one trip system, the required number of channels in the first column of Table 3.2.5 is "one." Note 8 of Table 3.2.5 is changed to be consistent with the Table 3.2.5 column heading change (i.e., change to "Required Channels"). This achieves consistency within the Technical Specifications, clarifies meaning, and reduces the potential for error by avoiding confusing terminology. The net result of this change is to re-define the minimum operability requirements of the control rod block instrumentation. These changes are more for clarification purposes and do not result in any actual change in plant operation. The operability requirements are technically equivalent, before and after the change. Therefore, this change only clarifies TS and has no negative impact on plant safety because technical requirements are not changed. The clarifications are intended to avoid confusion and the potential for erroneous plant operation with less than the required number of operable rod block channels. This change is acceptable because it maintains the level of protection provided by the current TS.

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
3	Table 3.2.5 and Table 4.2.5 contain Startup Range Monitor (Upscale and Detector Not Fully Inserted), Intermediate Range Monitor (Upscale, Downscale, and Detector Not Fully Inserted), and Average Power Range Monitor (Upscale and Downscale) rod block functions.	Remove the subject nuclear instrumentation rod block functions from TS Tables 3.2.5 and 4.2.5 and relocate these functions with their respective operability requirements and associated surveillance requirements to the VY TRM.
	Basis / Safety Assessment: The Source Range Monitor (SRM), Intermediate Range Monitor (IRM), and Average Power Range Monitor (APRM) control rod blocks function to prevent a control rod withdrawal error utilizing nuclear instrumentation signals during various plant modes of operation. No design basis accidents (DBA) or transient analyses take credit for rod block signals initiated by nuclear instrumentation. Comparing these control rod block functions to the screening criteria of 10CFR50.36 for inclusion in TS: 1. SRM, IRM, and APRM control rod block instrumentation are neither used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a DBA. 2. SRM, IRM, and APRM control rod block instrumentation do not monitor a process variable that is an initial condition of a DBA or transient. 3. SRM, IRM, and APRM control rod block instrumentation are not part of a primary success path in the mitigation of a DBA or transient. As discussed in Sections 3.5 and 6, and summarized in Table 4-1 (Items 135, 137, and 138) of NEDO-31466, loss of the SRM, IRM, and APRM control rod block functions was found to be a non-significant risk contributor to core damage frequency and offsite releases. VY also considers loss of the SRM, IRM, and APRM control rod block functions to be non-significant risk contributors. Based on these four criteria, the SRM, IRM, and APRM control rod block functions can be relocated to the TRM with no negative impact on plant safety. In addition, this is consistent with NUREG-1433.	

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
4	The current TS (including Tables 3.2.5 and 4.2.5) do not specify any requirements associated with the Inop function of the RBM.	RBM Inop operability and surveillance requirements are added to TS Tables 3.2.5 and 4.2.5, respectively. Table 3.2.5 A new operability requirement for the Inop function of the RBM is added to Table 3.2.5 and requires that a minimum of two instrument channels be operable when the reactor is in the run mode >30% rated thermal power. Existing Notes 7, 9 and 10 of Table 3.2.5 are also applicable to this function. (The bracket alongside existing Notes 10, 9 for the RBM is extended to bracket this added function.) There is no trip setting associated with the RBM Inop function. "(Notes 10, 9)" in the left column is changed to "(Notes 9 and 10)." Table 4.2.5 A requirement for performing a channel functional test of the RBM Inop function every three months is added to Table 4.2.5. There is no calibration applicable to the RBM Inop function.
	Basis / Safety Assessment: Adding requirements for the Inop function of the RBM is necessary to ensure that safety analysis assumptions regarding the RBM are maintained. Since the RBM is only required to be operable when the reactor is in the run mode >30% rated thermal power, the Inop function is only required to be operable during the same mode and conditions of operation. Notes 7, 9 and 10 of Table 3.2.5 are therefore applicable. The basis for requiring two operable channels is the same as discussed in Change #2 above. Changing the designation of "Notes 10, 9" to "Notes 9 and 10" is format related and represents a preferred manner of expression. Therefore, this is an administrative change. Verifying operability through a functional test every three months is acceptable since operating experience indicates that this test frequency provides reasonable assurance of operability. This change involves additional requirements not presently included in TS and is acceptable because it serves to ensure the operability of the RBM. The administrative change involving Notes 9 and 10 is merely a preferential change in format with no change in any technical requirements. Taken as a whole, this change is more restrictive on plant operation and has no negative impact on plant safety. In addition, the proposed surveillance frequency is consistent with NUREG-1433.	

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
5	Table 3.2.5 and Table 4.2.5 contain the Scram Discharge Volume (high water level) rod block function.	Remove the subject Scram Discharge Volume rod block function from TS Tables 3.2.5 and 4.2.5 and relocate this function with its operability requirements and associated surveillance requirements to the VY TRM.

Basis / Safety Assessment:

The Scram Discharge Volume (SDV) control rod block functions to prevent control rod withdrawals during power range operations, utilizing SDV water level signals to create the rod block if water accumulates in the SDV above a preset limit. 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 will occur at a predetermined value. Thus, the SDV water level rod block signal provides an opportunity for the operator to take corrective action to avoid a subsequent automatic scram. No DBA or transient analysis 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 also considers loss of the SDV control rod block function to be a non-significant risk contributor.

Based on these four criteria, the SDV control rod block functions can be relocated to the TRM with no negative impact on plant safety. This is consistent with NUREG-1433.

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
6	The current TS (including Tables 3.2.5 and 4.2.5) do not specify any requirements associated with the reactor mode switch in the shutdown position rod block function.	Reactor Mode Switch—Shutdown Position operability and surveillance requirements are added to TS Tables 3.2.5 and 4.2.5, respectively. Table 3.2.5 An operability requirement is added to Table 3.2.5 to require that two instrument channels be operable when this function is required to be operable. Note 12 is added to Table 3.2.5 to require this function to be operable when the reactor mode switch is in the shutdown position. Note 13 is added to Table 3.2.5 establishing the requirement to immediately suspend control rod withdrawal and immediately 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. There is no trip setting associated with this function since the channels are manually actuated. Table 4.2.5 An associated surveillance requirement is added to Table 4.2.5 that requires functional testing be performed every refueling outage. There is no calibration applicable to this trip function. Note 12 is added to Table 4.2.5 to clarify that trip system logic testing is not applicable to this function. Note 12 to Table 4.2.5 also requires that if the required surveillance frequency (every Refueling Outage) is not met, functional testing of the Reactor Mode Switch—Shutdown Position function shall be initiated within one hour after the reactor mode switch is placed in shutdown for the purpose of commencing a scheduled Refueling Outage.
	Basis / Safety Assessment: During shutdown conditions, no positive reactivity insertion events are analyzed because it is assumed that control rod withdrawal blocks are provided to prevent rod withdrawal. This change is acceptable since it ensures that shutdown rod block assumptions are valid and provides reasonable assurance to prevent inadvertent criticality. As such, this change enhances plant safety.	

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
7	Table 3.2.5 contains Notes 1, 2, 3, 4, and 11; and, Table 4.2.5 contains Note 6 that are only associated with functions that are being relocated to the TRM.	The Notes are relocated with their associated specifications (per Changes #3 and #5 above) 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 acceptable since it is administrative only and ensures consistency with Change #3 and Change #5; thus, there is no negative impact on plant safety.	
8	Note 7 to Table 3.2.5 states that the RBM trip may be bypassed when the reactor power is ≤30% of **rated**.	Note 7 of Table 3.2.5 is clarified to state that the RBM trip may be bypassed when the reactor power is ≤30% of **Rated Thermal Power**. The definitive term, "Rated Thermal Power," is substituted for the word, "rated," in Note 7 of Table 3.2.5.
	Basis / Safety Assessment: "Rated Thermal Power" is defined in VY TS Definitions (i.e., Specification 1.0.Q) and is a more precise term. This clarifies the conditions stated to avoid confusion and reduce the potential for error. This change is administrative in that it reflects terminology currently in use at VY, more accurately reflects the TS intent, yet does not change any TS technical requirement. This clarification is consistent with the design basis assumptions for the RBM. Because this change does not constitute a change in technical meaning of the TS, it is acceptable. As such, there is no negative impact on plant safety.	
9	Note 9, step a. to Table 3.2.5 requires a verification that the reactor is not operating on a limiting control rod pattern.	Note 9, step a. to Table 3.2.5 is revised by adding a parenthetical expression, as follows: "*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*"
	Basis / Safety Assessment: By adding the parenthetical expression, details are added to Note 9 to provide the control room operator with further understanding on the meaning of this required action. In addition, the Bases for Specification 3.3.B.6 are being revised and clarified to better define the meaning of a "limiting control rod pattern." The change to Note 9 does not change any technical requirements, but is administrative in nature. This change is acceptable since it provides clarifying detail to avoid confusion and the potential for error. As such, this administrative change has no negative impact on plant safety.	

Table 1
(continued)

Change #	Current Technical Specification	Proposed Change
10	The current TS (including Table 4.2.5) do not specify any requirements to ensure that the RBM Upscale (Flow Bias) function is not bypassed when >30% Rated Thermal Power.	A surveillance requirement is added to Table 4.2.5 to verify through calibration at a frequency of every three months that the RBM Upscale (Flow Bias) function is not bypassed when >30% Rated Thermal Power. Note 13 is added to Table 4.2.5 under the "Calibration" column for the RBM Upscale Function. Proposed Note 13 states: *Includes calibration of the RBM Reference Downscale function (i.e., RBM upscale function is not bypassed when >30% Rated Thermal Power).*
	Basis / Safety Assessment: APRM reference signals are used to automatically de-activate the RBM below a specified value, which must be ≤30% rated thermal power. Below this power level, the consequences of a rod withdrawal error event will not exceed the minimum critical power ratio safety limit, and therefore, the RBM is not required to be operable. The surveillance requirement is necessary to ensure that the RBM is operable when required (i.e., >30% rated thermal power). Plant procedures currently do require this surveillance at the proposed frequency. This change is acceptable since it ensures that the RBM is operable when required and is more restrictive on plant operations; therefore, there is no negative impact on plant safety.	
11	The TS Bases provide a discussion of functions that are being relocated to the TRM. In addition, other Bases could be clarified to provide a better understanding of the associated Specifications.	Those portions of the Bases associated with Specifications being relocated to the TRM will also be relocated to the TRM. Other Bases improvements are being made to clarify the associated Specifications.
	Basis / Safety Assessment: The Bases discussions that are only applicable to the Specifications being relocated to the TRM must be relocated to provide the proper context. They would have little meaning if left in the TS Bases. Other Bases changes are for clarity purposes and conformance to the changes being made to the associated Specifications. Bases do not establish actual requirements, and as such do not change technical requirements of the TS. Therefore, the changes are administrative in nature and have no negative impact on plant safety.	

Conclusion/Summary

Those TS requirements being relocated to the VY TRM are not essential to assure safe operation of the Vermont Yankee Nuclear Power Station, and do not meet the criteria of 10CFR50.36 for inclusion in TS. Control of the relocated requirements will be in accordance with 10CFR50.59, which ensures that NRC review and approval will be proposed for changes exceeding the applicable regulatory threshold.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

Docket No. 50-271
BVY 01-51

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 247

Control Rod Block Instrumentation

Determination of No Significant Hazards Consideration

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Description of amendment request:

The license amendment request would revise control rod block instrumentation requirements contained in Technical Specifications 2.1.B, Figure 2.1-1, Table 3.2.5 and Table 4.2.5.

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 inclusion in the Technical Specifications;
2. An imposition of more restrictive requirements to ensure operability that are driven by an effort for completeness and consistency with the BWR/4 Standard Technical Specifications; or
3. Administrative changes which add clarity, or are necessitated by relocating the associated Technical Specifications to the TRM.

The NRC staff has previously found, in other applications, the acceptability of relocating the identified trip 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, those provisions will administratively control subsequent revisions to these requirements.

Basis for no significant hazards determination:

Pursuant to 10CFR50.92, 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). These criteria require that the operation of the facility in accordance with the proposed amendment will not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The discussion below addresses each of these criteria and demonstrates that the proposed amendment does not constitute a significant hazard.

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 trip 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 the analyzed mitigation of design basis accident or transient events.

   More stringent requirements that ensure operability of equipment 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 change does 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 any accident. No new accident modes are created since plant operation 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.

   This change does not impact plant 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 assumptions or initial conditions, the margins of safety in the safety analyses are maintained. In addition, administrative changes that do not change technical requirements or meaning, and the imposition of more stringent requirements to ensure operability, have no negative impact on margins of safety. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Summary No Significant Hazards Consideration

Conclusion

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.

Docket No. 50-271
BVY 01-51

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 247

Control Rod Block Instrumentation

Marked-up Version of the Current Technical Specifications

BASES:

4.2 PROTECTIVE INSTRUMENTATION (Cont'd)

Since logic circuit tests result in the actuation of plant equipment, testing of this nature was done while the plant was shut down for refueling. In this way, the testing of equipment would not jeopardize plant operation.

This Specification is a periodic testing program which is based upon the overall testing of protective instrumentation systems, including logic circuits as well as sensor circuits. Table 4.2 outlines the test, calibration, and logic system functional test schedule for the protective instrumentation systems. The testing of a subsystem includes a functional test of each relay wherever practicable. The testing of each relay includes all circuitry necessary to make the relay operate, and also the proper functioning of the relay contacts. Testing of the automatic initiation inhibit switches verifies the proper operability of the switches and relay contacts. Functional testing of the inaccessible temperature switches associated with the isolation systems is accomplished remotely by application of a heat source to individual switches.

All subsystems are functionally tested, calibrated, and operated in their entirety.

11

A channel functional test is performed for the Reactor Mode Switch—Shutdown Position function to ensure that the entire channel will perform the intended function. The surveillance is only required to be performed once per operating cycle during refueling. The Refueling Outage frequency is based on the need to perform this surveillance under the conditions that apply during a plant outage. Operating experience has shown that this surveillance frequency is adequate to ensure functional operability. Note 12 of Table 4.2.5 specifies that if the required surveillance frequency of every Refueling Outage is not met, functional testing of the Reactor Mode Switch—Shutdown Position function shall be initiated within 1 hour after the reactor mode switch is placed in the Shutdown position for the purpose of commencing a scheduled Refueling Outage. This allows entry into the Shutdown mode when the surveillance requirement is not met.

INSERTS TO MARKED-UP PAGES

INSERT #1 (new Notes 12 and 13 to Table 3.2.5, Page 52)

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.

INSERT #2 (Bases 3.2, Page 77)

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.

INSERT #3 (Bases 3.2, Page 77)

During hot shutdown, cold shutdown, and refueling when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical with sufficient shutdown margin; 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.

INSERT #4 (Bases 3.3 & 4.3, Page 90)

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 value for APLHGR, LHGR, or MCPR).

Docket No. 50-271
BVY 01-51

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 247

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 included herein. The revised pages contain vertical lines in the margin indicating the areas of change.

Remove	Insert
8	8
9	9
11	11
16	16
20	20
51	51
52	52
69	69
74	74
77	77
78	78
80a	80a
90	90
122	122
259	259

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

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

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

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

B. Deleted

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

1.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

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BASES: 2.1 (Cont'd)

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

B. Deleted

C. Reactor Low Water Level Scram

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

D. Reactor Low Water Level ECCS Initiation Trip Point

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

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

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the operability of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant operation at any power level shall be permitted in accordance with Table 3.1.1. The system response time from the opening of the sensor contact up to and including the opening of the scram solenoid relay shall not exceed 50 milliseconds.

B. During operation at ≥25% Rated Thermal Power with the ratio of MFLPD to FRP greater than 1.0 either:

a. The APRM System gains shall be adjusted by the ratio given in Technical Specification 2.1.A.1, or

b. The power distribution shall be changed to reduce the ratio of MFLPD to FRP.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively

B. Once within 12 hours after ≥25% Rated Thermal Power and once a day during operation at ≥25% Rated Thermal Power thereafter, the maximum fraction of limiting power density and fraction of rated power shall be determined and the APRM system gains shall be adjusted by the ratio given in Technical Specification 2.1.A.1.a.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Required Channels		Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
(Notes 9 and 10)		Rod Block Monitor (RBM A/B)				
	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	
(Note 13)	2	Reactor Mode Switch - Shutdown Position (Note 12)				
(Note 8)	1	Trip System	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 the required number, 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	
Trip System Logic	Once/Operating Cycle	Once/Operating Cycle (Note 3)
Reactor Mode Switch - Shutdown Position	Every Refueling Outage (Note 12)	

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 testing is not applicable to this function. If the required surveillance frequency (every Refueling Outage) is not met, functional testing of the Reactor Mode Switch-Shutdown Position function shall be initiated within 1 hour after the reactor mode switch is placed in Shutdown for the purpose of commencing a scheduled Refueling Outage.

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 and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. The HPCI and RCIC steam supply pressure instrumentation is provided to isolate the systems when pressure may be too low to continue operation. These isolations are for equipment protection. However, they also provide a diverse signal to indicate a possible system break. These instruments are included in Technical Specifications because of the potential for possible system initiation failure if not properly tested. 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, cold shutdown, and refueling when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical with sufficient shutdown margin; 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

BASES: 3.2 (Cont'd)

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:

4.2 PROTECTIVE INSTRUMENTATION (Cont'd)

Since logic circuit tests result in the actuation of plant equipment, testing of this nature was done while the plant was shut down for refueling. In this way, the testing of equipment would not jeopardize plant operation.

This Specification is a periodic testing program which is based upon the overall testing of protective instrumentation systems, including logic circuits as well as sensor circuits. Table 4.2 outlines the test, calibration, and logic system functional test schedule for the protective instrumentation systems. The testing of a subsystem includes a functional test of each relay wherever practicable. The testing of each relay includes all circuitry necessary to make the relay operate, and also the proper functioning of the relay contacts. Testing of the automatic initiation inhibit switches verifies the proper operability of the switches and relay contacts. Functional testing of the inaccessible temperature switches associated with the isolation systems is accomplished remotely by application of a heat source to individual switches.

All subsystems are functionally tested, calibrated, and operated in their entirety.

A channel functional test is performed for the Reactor Mode Switch - Shutdown Position function to ensure that the entire channel will perform the intended function. The surveillance is only required to be performed once per operating cycle during refueling. The Refueling Outage frequeency is based on the need to perform this surveillance under the conditions that apply during a plant outage. Operating experience has shown that this surveillance frequency is adequate to ensure functional operability. Note 12 of Table 4.2.5 specifies that if the surveillance frequency of every Refueling Outage is not met, functional testing of the Reactor Mode Switch - Shutdown Position function shall be initiated within 1 hour after the reactor mode switch is placed in the Shutdown position for the purpose of commencing a scheduled Refueling Outage. This allows entry into the Shutdown mode when the surveillance requirement is not met.

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 provides a scram function in noncoincident configuration. 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 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.

3.6 LIMITING CONDITIONS FOR OPERATION

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Single Loop Operation

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

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

4.6 SURVEILLANCE REQUIREMENTS

3. The surveillance requirements of 4.6.F.1 and 4.6.F.2 do not apply to the idle loop and associated jet pumps when in single loop operation.

4. The baseline data required to evaluate the conditions in Specifications 4.6.F.1 and 4.6.F.2 shall be acquired each operating cycle. Baseline data for evaluating 4.6.F.2 while in single loop operation shall be updated as soon as practical after entering single loop operation.

include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, [1]/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

The dose assignment to various duty functions may be estimates based on Self-Reading Dosimeter (SRD), TLD or film badge measurement. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

B. Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the fifteenth of each month following the calendar month covered by the report. These reports shall include a narrative summary of operating experience during the report period which describes the operation of the facility.

C. Core Operating Limits Report

The core operating limits shall be established and documented in the Core Operating Limits Report (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. The Average Planar Linear Heat Generation Rates (APLHGR) for Specifications 3.11.A and 3.6.G.1a,

2. The K_f core flow adjustment factor for Specification 3.11.C.,

3. The Minimum Critical Power Ratio (MCPR) for Specifications 3.11.C and 3.6.G.1a,

4. The Linear Heat Generation Rates (LHGR) for Specifications 2.1.A.1a and 3.11.B, and

5. The Power/Flow Exclusion Region for Specifications 3.6.J.1a and 3.6.J.1b.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

Report, E. E. Pilat, "Methods for the Analysis of Boiling Water Reactors Lattice Physics," YAEC-1232, December 1980 (Approved by NRC SER, dated September 15, 1982).

[1]/ This tabulation supplements the requirements of 20.2206 of 10 CFR Part 20.