September 23, 1999

Mr. Robert J. Wanczyk
Acting Director of Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
Brattleboro, VT 05301

SUBJECT: REVISION OF TECHNICAL SPECIFICATION BASES PAGE - VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. MA6476)

Dear Mr. Wanczyk:

By letter dated August 31, 1999, you provided a revised technical specification (TS) bases page. To ensure that all persons on our mailing list are aware of the TS bases change and to ensure that the NRC and licensee have identical TS bases pages, the TS Bases page you provided is enclosed. The NRC staff did not perform an evaluation of the TS Bases pages and NRC concurrence with the page supplied should not be implied by this letter.

Sincerely,

Original signed by:

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

Docket No. 50-271

Enclosure: TS Bases Page 91

cc w/encl: See next page

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DATE	9/22/99		9/22/99		9/23/99					

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NAME	RPCroteau		TLClark		JClifford					
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Vermont Yankee Nuclear Power Station

cc:

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Putney, VT 05346-0566

BASES: 3.3 & 4.3 (Cont'd)

7. Periodic verification that the Scram Discharge Volume (SDV) drain and vent valves are maintained in the open position provides assurance that the SDV will be available to accept the water displaced from the control rod drives in the event of a scram.

C. Scram Insertion Times

The Control Rod System is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage. The limiting power transient is that resulting from a turbine stop valve closure with a failure of the Turbine Bypass System. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity safety limit.

The scram times for all control rods shall be determined during each operating cycle. The weekly control rod exercise test serves as a periodic check against deterioration of the Control Rod System and also verifies the ability of the control rod drive to scram. The frequency of exercising the control rods under the conditions of two or more control rods valved out of service provides even further assurance of the reliability of the remaining control rods.

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod (3x3) square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a $K_{eff} \leq 1.0$. Other repeating rod sequences with more rods withdrawn resulted in $K_{eff} \geq 1.0$. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the Control-Rod-Drive Hydraulic System. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

E. Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operation base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. One percent reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the Reactor System.