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BVY 09-074

December 9, 2009

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

SUBJECT: Revision of Technical Specifications Bases Pages
Vermont Yankee Nuclear Power Station
Docket No. 50-271
License No. DPR-28

REFERENCES:

1. Letter, USNRC to Entergy, "Vermont Yankee Nuclear Power Station - Issuance of Amendment RE: Relocation of Reactor Building Crane Technical Specification (TAC NO. MD9725)," NVY 09-077, dated July 13, 2009

2. Letter, USNRC to Entergy, "Vermont Yankee Nuclear Power Station - Issuance of Amendment RE: Testing and Inspection of Suppression Chamber-to-Drywell Vacuum Breakers (TAC NO. ME0767)," NVY 09-073, dated July 6, 2009

Dear Sir or Madam:

This letter provides revised Vermont Yankee Nuclear Power Station (VY) Technical Specification (TS) Bases pages. The TS Bases were revised in conjunction with Amendments to Operating License DPR-28 issued in References 1 and 2.

These changes, processed in accordance with our Technical Specification Bases Control Program (TS 6.7.E), were determined not to require prior NRC approval. The revised Bases pages are provided in Attachment 1 for your information and for updating and inclusion with your copy of VY Technical Specifications. No NRC action is required in conjunction with this submittal.

There are no new regulatory commitments being made in this submittal.

Should you have any questions concerning this submittal, please contact me at 802-451-3304.

Sincerely,

[DJM/PLC]

Attachment: 1. Revised Technical Specifications Bases Pages

cc: Mr. Samuel J. Collins
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Attachment 1

Vermont Yankee Nuclear Power Station

Revised Technical Specifications Bases Pages

BASES: 4.7 (Cont'd)

The maximum allowable test leak rate at the peak accident pressure of 44 psig (La) is 0.80 weight % per day. The maximum allowable test leak rate at the retest pressure of 24 psig (Lt) has been conservatively determined to be 0.59 weight percent per day. This value was verified to be conservative by actual primary containment leak rate measurements at both 44 psig and 24 psig upon completion of the containment structure.

As most leakage and deterioration of integrity is expected to occur through penetrations, especially those with resilient seals, a periodic leak rate test program of such penetration is conducted at the peak accident pressure of 44 psig to insure not only that the leakage remains acceptably low but also that the sealing materials can withstand the accident pressure.

The Primary Containment Leak Rate Testing Program is based on Option B to 10CFR50, Appendix J, for development of leak rate testing and surveillance schedules for reactor containment vessels.

Surveillance of the suppression Chamber-Reactor Building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak-tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. Operability testing is performed in conjunction with Specification 4.6.E. Calibrations are performed during the refueling outages; this frequency being based on equipment quality, experience, and engineering judgment.

The ten (10) drywell-suppression vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified.

The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 0.12 ft^2. This is equivalent to one vacuum breaker open by three-eighths of an inch (3/8") as measured at all points around the circumference of the disk or three-fourths of an inch (3/4") as measured at the bottom of the disk when the top of the disk is on the seat. Since these valves open in a manner that is purely neither mode, a conservative allowance of one-half inch (1/2") has been selected as the maximum permissible valve opening. Assuming that permissible valve opening could be evenly divided among all ten vacuum breakers at once, valve open position assumed to indication for an individual valve must be activated less than fifty-thousandths of an inch (0.050") at all points along the seal surface of the disk. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a nonseated valve.

BASES: 4.7 (Cont'd)

At the end of each refueling cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure and held constant. The 2 psig set point will not be exceeded. The subsequent suppression chamber pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed. If the drywell pressure can be increased by 1 psi over the suppression chamber the rate of change of the suppression chamber pressure must not exceed a rate equivalent to the rate of leakage from the drywell through a 1-inch orifice. In the event the rate of change exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised in accordance with Specification 4.6.E, following termination of discharge of steam into the suppression chamber from the safety/relief valves and following any operation that causes the vacuum breakers to open. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights are designed to function as follows:

Full Closed (Closed to $\leq$0.050" open)	2 White - On
Open (>0.050" open to full open)	2 White - Off

Experience has shown that a weekly measurement of the oxygen concentration in the primary containment assures adequate surveillance of the primary containment atmosphere.

B. and C. Standby Gas Treatment System and Secondary Containment System

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 0.15 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leakage tightness of the reactor building, and performance of the standby gas treatment system. The testing of reactor building automatic ventilation system isolation valves in accordance with Technical Specification 4.6.E demonstrates the operability of these valves. In addition, functional testing of initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

BASES: 3.12 & 4.12 (Cont'd)

E. The intent of this specification is to permit the unloading of a portion of the reactor core for such purposes as inservice inspection requirements, examination of the core support plate, control rod, control rod drive maintenance, etc. This specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in the Bases for Specification 3.12.A. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

One method available for unloading or reloading the core is the spiral unload/reload. Spiral reloading and unloading encompass reloading or unloading a cell on the edge of a continuous fueled region (the cell can be reloaded or unloaded in any sequence.) The pattern begins (for reloading) and ends (for unloading) around a single SRM. The spiral reloading pattern is the reverse of the unloading pattern, with the exception that two diagonally adjacent bundles, which have previously accumulated exposure in-core, and placed next to each of the four SRMs before the actual spiral reloading begins. The spiral reload can be to either the original configuration or a different configuration.

Additionally, at least 50% of the fuel assemblies to be reloaded into the core shall have previously accumulated a minimum exposure of 1000 Mwd/T to ensure the presence of a minimum neutron flux as described in Bases Section 3.12.B.

F. The intent of this specification is to assure that the reactor core has been shut down for at least 24 hours following power operation and prior to fuel handling or movement. The safety analysis for the postulated refueling accident assumed that the reactor had been shut down for 24 hours for fission product decay prior to any fuel handling which could result in dropping of a fuel assembly.

G. Deleted

H. The Spent Fuel Pool Cooling System is designed to maintain the pool water temperature below 125°F during normal refueling operations. If the reactor core is completely discharged, the temperature of the pool water may increase to greater than 125°F. The RHR System supplemental fuel pool cooling may be used under these conditions to maintain the pool water temperature less than 150°F.