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August 31, 2000
BVY 00-78

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)
Revision of Technical Specification Bases Pages

This letter provides revised Technical Specification (TS) Bases pages.

Bases page 29 has been revised to correct references to FSAR sections. This editorial change provides a general reference to the appropriate FSAR chapter for a more detailed description of the Neutron Monitoring Systems.

Bases page 80 has been revised to include the correct date for a referenced General Electric Topical Report. This is an editorial change to correct a typographical error.

Bases page 164 has been revised to reflect modifications to the suppression chamber downcomers that were performed during our 1998 refueling outage. The modifications involved cutting the downcomers to reduce submergence, and thus decreasing postulated blowdown loads on the suppression chamber internal structures and shell.

These changes to the TS Bases have been determined to not involve an unreviewed safety question in accordance with 10CFR50.59 and therefore do not require prior NRC approval.

Should you have any questions concerning this matter, please contact Mr. Jeffrey T. Meyer at (802) 258-4105.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Gautam Sen

Gautam Sen
Licensing Manager

Attachments

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

A001

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Technical Specification Bases Revision

Marked-up Pages

BASES:

3.1 Reactor Protection System

The reactor protection system automatically initiates a reactor scram to:

1. preserve the integrity of the fuel barrier;
2. preserve the integrity of the primary system barrier; and
3. minimize the energy which must be absorbed, and prevent criticality following a loss of coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance, testing, or calibration. The basis for the allowable out-of-service times is provided in GE Topical Report NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," March 1988.

The reactor protection system is of the dual channel type. The system is made up of two independent logic channels, each having three subsystems of tripping devices. One of the three subsystems has inputs from the manual scram push buttons and the reactor mode switch. Each of the two remaining subsystems has an input from at least one independent sensor monitoring each of the critical parameters. The outputs of these subsystems are combined in a 1 out of 2 logic; i.e., an input signal on either one or both of the subsystems will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both logic channels is required to produce a reactor scram.

The required conditions when the minimum instrument logic conditions are not met are chosen so as to bring station operation promptly to such a condition that the particular protection instrument is not required; or the station is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operating conditions.

When the minimum requirements for the number of operable or operating trip system and instrumentation channels are satisfied, the effectiveness of the protection system is preserved; i.e., the system can tolerate a single failure and still perform its intended function of scramming the reactor.

Three APRM instrument channels are provided for each protection trip system to provide for high neutron flux protection. APRM's A and E operate contacts in a trip subsystem, and APRM's C an E operate contacts in the other trip subsystem. APRM's B, D, and F are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration without changing the minimum number of channels required for inputs to each trip system. Additional IRM channels have also been provided to allow bypassing of one such channel. ~~IRM assignment to the bypass switches is described on FSAR Figure 7.5-9 and in FSAR Section 7.5.5.4.~~

For a description of the Neutron Monitoring Systems, see FSAR Section 7.5

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specification 2.1.

BASES:

4.2 PROTECTIVE INSTRUMENTATION

The Protective Instrumentation Systems covered by this Specification are listed in Table 4.2. Most of these protective systems are composed of two or more independent and redundant subsystems which are combined in a dual-channel arrangement. Each of these subsystems contains an arrangement of electrical relays which operate to initiate the required system protective action.

The relays in a subsystem are actuated by a number of means, including manually-operated switches, process-operated switches (sensors), bistable devices operated by analog sensor signals, timers, limit switches, and other relays. In most cases, final subsystem relay actuation is obtained by satisfying the logic conditions established by a number of these relay contacts in a logic array. When a subsystem is actuated, the final subsystem relay(s) can operate protective equipment, such as valves and pumps, and can perform other protective actions, such as tripping the main turbine-generator unit.

With the dual-channel arrangement of these subsystems, the single failure of a ready circuit can be tolerated because the redundant subsystem or system (in the case of high pressure coolant injection) will then initiate the necessary protective action. If a failure in one of these circuits occurs in such a way that an action is taken, the operator is immediately alerted to the failure. If the failure occurs and causes no action, it could then remain undetected, causing a loss of the redundancy in the dual-channel arrangement. Losses in redundancy of this nature are found by periodically testing the relay circuits and contacts in the subsystems to assure that they are operating properly.

The surveillance test interval for the instrumentation channel functional tests are once/three months for most instrumentation. The allowable out-of-service times and surveillance interval is based on the following NRC approved licensing topical reports:

1. GE Topical Report NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," March 1988.

2. GE Topical Report NEDC-30851P-A, Supplement 1 "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

3. GE Topical Report NEDC-30851P-A, Supplement 2 "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," ~~July 1986.~~ March 1989

4. GE Topical Report NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

5. GE Topical Report NEDC-30936P-A, Parts 1 and 2, "BWR Owners Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," December 1988.

6. GE Topical Report GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

7. GE Topical Report GENE-770-06-2-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

BASES: 3.7 (Cont'd)

In conjunction with the Mark I Containment Long-Term Program, a plant unique analysis was performed (see Vermont Yankee letter, dated April 27, 1984, transmitting Teledyne Engineering Services Company Reports, TR-5319-1, Revision 2, dated November 30, 1983 and TR-5319-2, Revision 0) which demonstrated that all stresses in the suppression chamber structure, including shell, external supports, vent system, internal structures, and attached piping meet the structural acceptance criteria of NUREG-0661. The maintenance of a drywell-suppression chamber differential pressure of 1.7 psid and a suppression chamber water level corresponding to a downcomer submergence ~~range~~ of ~~4.29 to~~ less than 4.54 ft. will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

Using a 50°F rise (Section 5.2.4 FSAR) in the suppression chamber water temperature and a minimum water volume of 68,000 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 100°F will assure that the 170°F limit is not approached.

In the event a relief valve inadvertently opens or sticks open, operating procedures define the action to be taken. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Double isolation valves are provided on lines which penetrate the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section 5.2 of the FSAR.

Manual primary containment isolation valves that are required to be closed by the definition of Primary Containment Integrity may be opened intermittently under administrative controls. These controls consist of stationing a dedicated operator, with whom Control Room communication is immediately available, in the immediate vicinity of the valve controls. In this way, the penetration can be rapidly isolated when a need for primary containment isolation is indicated.

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Re-typed Pages

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