February 26, 98

Mr. Donald A. Reid
Senior Vice President, Operations
Vermont Yankee Nuclear Power Corporation
Ferry Road
Brattleboro, VT 05301

SUBJECT: ISSUANCE OF AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-28, VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. M99264)

Dear Mr. Reid:

The Commission has issued the enclosed Amendment No. 152 to Facility Operating License DPR-28 Vermont Yankee Nuclear Power Station. The amendment changes the Technical Specifications (TSs) in response to your application dated July 11, 1997, as supplemented by letters dated November 21, December 22, 1997, and February 6, 1998.

The amendment revises Technical Specifications (TSs) 3.7/4.7 and their associated Bases to incorporate Option B of Appendix J to 10 CFR 50, and editorial changes to TS Table 4.7.2.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Kahtan N. Jabbour, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 152 to DPR-28
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\JABBOUR\VYM 99264.AMD

*see previous concurrence

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DATED: February 26, 1998

AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-28 VERMONT YANKEE ATOMIC POWER STATION

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 152
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for the amendment by the Vermont Yankee Nuclear Power Corporation (the licensee) dated July 11, 1997, as supplemented November 21, December 22, 1997, and February 6, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

   (B) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 152, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Cecil O. Thomas

Cecil O. Thomas, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specification

Date of Issuance: February 26, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 152

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

Remove	Insert
147	147
156	156
157	157
158	158
159	159
160	160
161	161
168	168
279	279

3.7 LIMITING CONDITIONS FOR OPERATION

e. Minimum Water Volume - 68,000 cubic feet

f. Maximum Water Volume - 70,000 cubic feet

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

3. If a portion of a system that is considered to be an extension of primary containment is to be opened, isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve or blind flange.

4. Whenever primary containment is required, the leakage from any one main steam line isolation valve shall not exceed 15.5 scf/hr at 44 psig (P_a).

4.7 SURVEILLANCE REQUIREMENTS

2. The primary containment integrity shall be demonstrated as required by the Primary Containment Leak Rate Testing Program (PCLRTP).

3. (Blank)

4. The leakage from any one main steam line isolation valve shall not exceed 11.5 scf/hr at 24 psig (Pt). Repair and retest shall be conducted to insure compliance.

3.7 LIMITING CONDITIONS FOR OPERATION

5. Core spray and LPCI pump lower compartment door openings shall be closed at all times except during passage or when reactor coolant temperature is less than 212°F.

D. <u>Primary Containment Isolation Valves</u>

1. During reactor power operating conditions all containment isolation valves and all instrument line flow check valves shall be operable except as specified in Specification 3.7.D.2.

4.7 SURVEILLANCE REQUIREMENTS

5. The core spray and LPCI lower compartment openings shall be checked closed daily.

D. <u>Primary Containment Isolation Valves</u>

1. Surveillance of the primary containment isolation valves should be performed as follows:

   a. The operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and the closure times specified in Table 4.7.2 at least once per operating cycle.

   b. Operability testing of the primary containment isolation valves shall be performed in accordance with Specification 4.6.E.

   c. At least once per quarter, with the reactor power less than 75 percent of rated, trip all main steam isolation valves (one at a time) and verify closure time.

   d. At least twice per week, the main steam line isolation valves shall be exercised by partial closure and subsequent reopening.

3.7 LIMITING CONDITIONS FOR OPERATION

2. In the event any containment isolation valve becomes inoperable, reactor power operation may continue provided at least one containment isolation valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specifications 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

4.7 SURVEILLANCE REQUIREMENTS

2. Whenever a containment isolation valve is inoperable, the position of at least one other valve in each line having an inoperable valve shall be logged daily.

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TABLE 4.7.2

PRIMARY CONTAINMENT ISOLATION VALVES

Isolation Group (1)	Valve Identification	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A, D & 2-86A, D)	4	4	5 (Note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SC
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9)		1	10	Closed	SC
3	Drywell Air Purge Inlet (16-19-8)		1	10	Closed	SC
3	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed*	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed*	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Open	GC

* Valves 16-19-7 and 16-19-7A shall have stops installed to limit valve opening to 50° or less.

TABLE 4.7.2
(Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

Isolation Group (1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
3	Exhaust to Standby Gas Treatment System (16-19-6)		1	10	Open	GC
3	Containment Purge Supply (16-19-23)		1	10	Closed	SC
3	Containment Makeup Supply (16-20-22A)		1	NA	Closed	SC
3	Containment Makeup Supply (16-20-20, 16-20-22B)		2	5	Open	GC
5	Reactor Cleanup System (12-15, 12-18)	1	1	25	Open	GC
6	HPCI (23-15, 23-16)	1	1	55	Open	GC
6	RCIC (13-15, 13-16)	1	1	20	Open	GC
	Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B)		2	NA	Closed	SC
	Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B)		2	NA	Closed	Process
3	Containment Air Sampling (VG 23, VG 26, 109-76A&B)		4	5	Open	GC
	Feedwater Check Valves (V2-27A, -96A, -28A, -28B)			NA	Open	Process

TABLE 4.7.2
(Cont'd)

PRIMARY CONTAINMENT ISOLATION VALVES

Isolation Group (1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
2	RHR Return to Suppression Pool (10-39A, B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A, B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A, B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A, B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A, B)		2	20	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Proc.
*	Hydrogen Monitoring (109-75 A, 1-4; 109-75 B-D, 1-2) Sampling Valves - Inlet		10	NA	NA	NA
*	Hydrogen Monitoring (VG-24, 25, 33, 34)		4	NA	NA	NA

* These valves are remote manual sampling valves which do not receive an isolation signal. Only one valve in each line is required to be operable.

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 will be 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. Inspections and 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. Also it is inspected to assure that it closes freely and operates properly.

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.

4. An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

5. An evaluation of the change, which shows the expected maximum exposures to member(s) of the public at the site boundary and to the general population that differ from those previously estimated in the license application and amendments thereto;

6. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

7. An estimate of the exposure to plant operating personnel as a result of the change; and

8. Documentation of the fact that the change was reviewed and found acceptable by PORC.

B. Shall become effective upon review and acceptance by PORC and approval by the Plant Manager.

6.15 Primary Containment Leak Rate Testing Program

A program shall be established to implement the leak rate testing of the primary containment as required by 10CFR50.54(o) and 10CFR50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, entitled "Performance Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 44 psig.

The maximum allowable primary containment leak rate, La, at Pa, shall be 0.8% of primary containment air weight per day.

Leak rate acceptance criteria are:

1. Primary containment leak rate acceptance criterion $\leq$ 1.0 La.

2. The as-left primary containment integrated leak rate test (Type A test) acceptance criterion is $\leq$ 0.75 La.

3. The combined local leak rate test (Type B and C tests) acceptance criterion is $\leq$ 0.60 La, calculated on a maximum pathway basis, prior to entering a mode of operation where containment integrity is required.

4. The combined local leak rate test (Type B and C tests) acceptance criterion is $\leq$ 0.60 La, calculated on a minimum pathway basis, at all times when primary containment integrity is required.

5. Airlock overall leak rate acceptance criterion is $\leq$ 0.10 La when tested at $\geq$ Pa.

The provision of the Definition (1.0.Y) for Surveillance Frequency does not apply to the test frequencies specified in the Primary Containment Leak Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 152 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

On September 12, 1995, the U.S. Nuclear Regulatory Commission (NRC) approved issuance of a revision to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors" which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The NRC added Option B, "Performance-Based Requirements," to allow licensees to voluntarily replace the prescriptive testing requirements of 10 CFR Part 50, Appendix J, with testing requirements based on both overall performance and the performance of individual components.

By application dated July 11, 1997, as supplemented by letters dated November 21, December 22, 1997, and February 6, 1998, Vermont Yankee Nuclear Power Corporation (the licensee) requested changes to the Technical Specifications (TS) for the Vermont Yankee Nuclear Power Station. The November 21, December 22, 1997, and February 6, 1998, letters did not change the initial proposed no significant hazards determination. The proposed changes would permit implementation of 10 CFR Part 50, Appendix J, Option B, and reference Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, which specifies a method acceptable to the NRC for complying with Option B.

2.0 BACKGROUND

Compliance with 10 CFR Part 50, Appendix J, provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TS and Bases. The allowable leakage rate is determined so that the leakage rate assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. 10 CFR Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of this revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

The RG 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, was developed as a method acceptable to the NRC staff for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the NRC staff for complying with Option B with four exceptions which are described therein.

Option B requires that RG 1.163 or another implementation document used by a licensee to develop a performance-based leakage testing program must be included, by general reference, in the plant TS. The licensee has referenced RG 1.163 in the proposed Vermont Yankee TS.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to five years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TS to implement Option B. After some discussion, the staff and NEI agreed on final TS which were transmitted to NEI in a letter dated November 2, 1995. These TS are to serve as a model for licensees to develop plant-specific TS in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of, or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

The licensee's July 11, November 21, December 22, 1997, and February 6, 1998, letters to the NRC propose to establish a "Primary Containment Leakage Rate Testing Program" and propose to add this program to the TS. The program references RG 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, which specifies methods acceptable to the NRC for complying with Option B. The licensee proposes changes to existing TSs 3.7.A.3, 3.7.A.4, 4.7.A.2, 4.7.A.3, 4.7.A.4, 3.7.D.1, 3.7.D.2, 4.7.D.2, Table 4.7.1, Table 4.7.2.a, and Table 4.7.2.b, and the addition of the "Primary Containment Leakage Rate Testing Program" as TS 6.15. Corresponding bases were also modified.

Option B permits a licensee to choose Type A; or Type B and C; or Type A, B and C; testing to be done on a performance basis. The licensee has elected to perform Type A, B, and C testing on a performance basis.

The TS changes proposed by the licensee are in compliance with the requirements of Option B and are consistent with the guidance of RG 1.163. Further, the proposed TS changes generally conform to the model TS guidance provided in the NRC letter to NEI dated November 2, 1995, despite the different format of the licensee's current TSs. The specific TS changes are discussed below in 3.1.

Additionally, the licensee has proposed related TS changes which go beyond the scope of the conversion to Option B; these are discussed in 3.2, below.

3.1 OPTION B TS CHANGES

Current TS 3.7.A.3 is being deleted, and TS 4.7.A.2 is being revised, to remove details about the testing program from these sections and to replace those details with a reference to the Primary Containment Leakage Rate Testing Program. This is consistent with the model TS guidance and is acceptable.

Current TSs 3.7.A.4 and 4.7.A.4 are being revised to remove a requirement that the leakage rate from any one containment isolation valve (CIV) shall not exceed 5 percent of the maximum allowable containment leak rate. This limitation goes beyond even the requirements of Option A of Appendix J, and beyond the requirements of Option B or the provisions of RG 1.163. As mentioned in section 2.0 above, the licensee must establish individual administrative limits for each CIV, but they are not TS requirements. Therefore, the staff finds this change to be acceptable.

Current TSs 3.7.D.1, 3.7.D.2, and 4.7.D.2 are being revised to remove references to TS tables that list CIVs. The revised TS will apply to all CIVs, rather than just those listed in TS tables. Further, TS Table 4.7.1, "Penetrations and Seals Subject to Type B Testing," is being deleted, and Tables 4.7.2.a and 4.7.2.b, which are tables listing CIVs, are being revised so that they no longer indicate which valves are Type C tested. This is consistent with the provisions of GL 91-08, "Removal of Component Lists From Technical

Specifications," dated May 6, 1991, and with the model TS for Option B implementation, which assume that component lists have already been removed from the plant's TS. Therefore, the staff finds these changes to be acceptable.

A new TS 6.15, "Primary Containment Leakage Rate Testing Program," is being added to describe the program. It is consistent with the model TS, except that the licensee is proposing additional words, beyond the model TS, for the leakage rate acceptance criteria, to reflect these acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates. The model Bases for TS 3.6.1.1.1, state:

> Reviewer's Note: Regulatory Guide 1.163 and NEI 94-01 include acceptance criteria for as-left and as-found Type A leakage rates and combined Type B and C leakage rates, which may be reflected in the Bases.

As an extension of this concept, the licensee is proposing to put the additional words into the TS itself. The staff has reviewed these additional words and finds that they are consistent with RG 1.163 and NEI 94-01, and are therefore acceptable.

Proposed TS 6.15 also deviates from the model TS in that it does not state a separate air lock leakage rate testing acceptance criterion for reduced-pressure door seal tests; it only gives an acceptance criterion for the overall, full-pressure test. At this plant, all air lock tests are performed at full pressure. The full-pressure test is a better test, in that it encompasses the entire air lock, rather than just the door seals, and the higher test pressure results in a more accurate leakage rate measurement and challenges the components more than a reduced-pressure test. Therefore, the staff finds the proposed air lock acceptance criterion to be acceptable.

3.2 <u>RELATED TS CHANGES</u>

TS 4.7.A.3

Current TS 4.7.A.3 requires that, before opening or breaking a closed piping system outside containment, which is connected to certain CIVs that are not Type C tested (in other words, a system that acts as an extension of containment), the isolation valves bounding the opening shall be Type C tested or else a blind flange shall be installed on the opening. The licensee proposes to delete this surveillance requirement and replace it with a new Limiting Condition For Operation (LCO) which says:

> If a portion of a system that is considered to be an extension of primary containment is to be opened, isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve or blind flange.

First, the staff agrees that this is more properly an LCO than a surveillance requirement. Second, the Improved Standard TS and the model TS for Option B implementation both allow open penetration flow paths (usually due to inoperable CIVs) to be isolated by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. Third, the licensee states that the closed and deactivated automatic valves, closed manual valves, or blind flanges will be containment isolation devices and will be leak rate tested in accordance with the Primary Containment Leakage Rate Testing Program. Therefore, the staff finds the proposed TS to be essentially equivalent to the existing TS and to represent no significant increase in risk to public health and safety; thus, it is acceptable.

Table 4.7.2 Editorial Change

Currently, TS Table 4.7.2 indicates that two of the containment purge valves are normally open to support drywell-to-suppression chamber differential pressure control and suppression pool water level. In a 1978 letter, the licensee described their procedure, including the use of a pumpback system to maintain the differential pressure. Subsequently, the licensee changed the method and described a different purge/vent valve line-up in a 1982 letter. The staff accepted their proposal in a letter dated May 3, 1982, but the TS table was not amended to reflect the fact that the two purge valves shown as normally open would instead be normally closed, and two other, smaller valves (not currently listed in the table) would be normally open. Thus, the proposed change is editorial in nature.

All four valves are CIVs and receive automatic containment isolation signals to close in the event of an accident. The two valves, now to be normally open, are smaller and will close faster than the two valves that were formerly open, so public risk will be slightly reduced.

On the basis given above, the staff finds the proposed changes to TS Table 4.7.2 to be acceptable.

3.3 SUMMARY

In summary, the staff has reviewed the changes to the TS and associated Bases proposed by the licensee, for Option B implementation, and finds that they are in compliance with the requirements of Appendix J, Option B, and are consistent with the guidance of RG 1.163, and are therefore acceptable. Further, the staff finds the additional changes discussed in section 3.2 above to be acceptable on the bases discussed therein.

4.0 STATE CONCLUSION

In accordance with the Commission's regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (62 FR 45465). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

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