April 2, 2001

Mr. Michael A. Balduzzi
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
185 Old Ferry Road
P.O. Box 7002
Brattleboro, VT 05302-7002

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - RELIEF REQUEST FOR EXCESS FLOW CHECK VALVE TESTING IN THE PUMP AND VALVE INSERVICE TEST PROGRAM (TAC NO. MB0415)

Dear Mr. Balduzzi:

By letters dated October 31, 2000, and January 25, 2001, Vermont Yankee Nuclear Power Corporation (VYNPC) submitted a relief request (RR-V19) for Vermont Yankee Nuclear Power Station (VY). VYNPC requested relief for excess flow check valves (EFCVs) from the American Society of Mechanical Engineers' (ASME) Code inservice tests that are required to be performed every refueling outage, and from the bienniel requirement to verify that the valve position is accurately indicated.

EFCVs are required by VY Technical Specification 4.6.E.2 and Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a to be inservice tested in accordance with the ASME Code. The proposed alternative described in Relief Request RR-V19 would relax the surveillance frequency by limiting the number of tests to a "representative sample" of EFCVs during each 18-month refueling outage, such that each EFCV will be tested at least once every 10 years (nominal). The justification for the relief request is based on General Electric Nuclear Energy (GE) Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" dated June 2000. The topical report provided: (1) an estimate of steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close, and (2) an assessment of the radiological consequences of such a release. The staff reviewed the licensee's submittals for applicability to GE Topical Report NEDO-32977-A and conformance with approved staff guidance regarding radiological dose assessment, EFCV failure rate and release frequency, and the proposed failure feedback mechanism and corrective action program.

The staff has completed its review of Relief Request RR-V19 and pursuant to 10 CFR 50.55a(a)(3)(i), Relief Request RR-V19 is authorized for use on the basis that the proposed alternative provides an acceptable level of quality and safety. This completes the staff's review and closes TAC No. MB0415.

Sincerely,

/RA/ R.B. Ennis for

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosure: Safety Evaluation

cc w/encl: See next page

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* Input received on 3/9/01.
No major changes were made.
**See previous concurrence

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cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. David R. Lewis
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037-1128

Ms. Christine S. Salembier, Commissioner
Vermont Department of Public Service
112 State Street
Montpelier, VT 05620-2601

Mr. Michael H. Dworkin, Chairman
Public Service Board
State of Vermont
112 State Street
Montpelier, VT 05620-2701

Chairman, Board of Selectmen
Town of Vernon
P.O. Box 116
Vernon, VT 05354-0116

Mr. Richard E. McCullough
Operating Experience Coordinator
Vermont Yankee Nuclear Power Station
P.O. Box 157
Governor Hunt Road
Vernon, VT 05354

G. Dana Bisbee, Esq.
Deputy Attorney General
33 Capitol Street
Concord, NH 03301-6937

Chief, Safety Unit
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Deborah B. Katz
Box 83
Shelburne Falls, MA 01370

Mr. Raymond N. McCandless
Vermont Department of Health
Division of Occupational
and Radiological Health
108 Cherry Street
Burlington, VT 05402

Mr. Gautam Sen
Licensing Manager
Vermont Yankee Nuclear Power
Corporation
185 Old Ferry Road
P.O. Box 7002
Brattleboro, VT 05302-7002

Resident Inspector
Vermont Yankee Nuclear Power Station
U. S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, VT 05354

Director, Massachusetts Emergency
Management Agency
ATTN: James Muckerheide
400 Worcester Rd.
Framingham, MA 01702-5399

Jonathan M. Block, Esq.
Main Street
P. O. Box 566
Putney, VT 05346-0566

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST FOR EXCESS FLOW CHECK VALVE TESTING FREQUENCY

VERMONT YANKEE NUCLEAR POWER STATION

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

1.0 INTRODUCTION

Title 10 of the *Code of Federal Regulations* (10 CFR) section 50.55a, requires that inservice testing (IST) of certain American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 pumps and valves are performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (the Code) and applicable addenda, except where alternatives have been authorized or relief has been requested by the licensee and granted by the Commission pursuant to Sections (a)(3)(i), (a)(3)(ii), or (f)(6)(i) of 10 CFR 50.55a. In proposing alternatives or requesting relief, the licensee must demonstrate that: (1) the proposed alternatives provide an acceptable level of quality and safety; (2) compliance would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety; or (3) conformance is impractical for its facility. Title 10 of the *Code of Federal Regulations,* section 50.55a authorizes the Commission to approve alternatives and to grant relief from ASME Code requirements upon making the necessary findings. Guidance related to the development and implementation of IST programs is given in Generic Letter (GL) 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," issued April 3, 1989, and its Supplement 1 issued April 4, 1995. Also see NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," and NUREG/CR-6396, "Examples, Clarifications, and Guidance on Preparing Requests for Relief from Pump and Valve Inservice Testing Requirements."

The 1989 Edition of the ASME Code is the applicable code of record for the third 10-year interval IST program at the Vermont Yankee Nuclear Power Station (VY). Subsection IWV of the 1989 Edition, which gives the requirements for IST of valves, references Part 10 of the American National Standards Institute/ASME *Operations and Maintenance Standards* (OM-10) as the rules for IST of valves. OM-10 replaces specific requirements in previous editions of Section XI, Subsection IWV, of the ASME Code. Subsection IWP of the 1989 Edition, which gives the requirements for IST of pumps, references Part 6 of the American National Standards Institute/ASME *Operations and Maintenance Standards* (OM-6) as the rules for IST of pumps. OM-6 replaces specific requirements in previous editions of Section XI, Subsection IWP, of the ASME Code.

By letters dated October 31, 2000, and January 25, 2001, Vermont Yankee Nuclear Power Corporation (VYNPC) submitted a relief request (RR-V19) for VY. VYNPC requested relief for various excess flow check valves (EFCV) from the ASME Code inservice tests that are required to be performed every refueling outage as specified in OM-10 Code, Paragraph 4.3.2.2, and from the bienniel requirements (Paragraph 4.2 of OM-10 Code) of verifying that the valve position is accurately indicated. The staff has completed its review of relief request RR-V19 and is providing the following evaluation.

Excess Flow Check Valves Included in Relief Request RR-V19

SL-13-55A,B,C,D	SL-14-31A,B
SL-2-62A,B,C,D	SL-2-64A,B,C,D
SL-2-73A,B,C,D,E,F,G,H	SL-2-2-7A,B
SL-2-2-8A,B	SL-2-3-11
SL-2-3-13A,B	SL-2-3-15A,B
SL-2-3-17A,B	SL-2-3-19A,B
SL-2-3-21A,B,C,D	SL-2-3-23A,B,C,D
SL-2-3-25	SL-2-3-27
SL-2-3-31A,B,C,D,E,F,G,H,I,J,K,L,M,N,P,Q	
SL-2-3-33	SL-2-3-35
SL-2-305A,B	SL-23-37A,B,C,D

2.0 EVALUATION

EFCVs are installed on boiling-water reactor (BWR) instrument lines to limit the release of fluid in the event of an instrument line break. Examples of EFCV installations include reactor pressure vessel level and pressure instrumentation, main steam line flow instrumentation, recirculation pump suction pressure, and reactor core isolation cooling steam line flow instrumentation. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post loss-of-coolant accident (LOCA) conditions.

The standard technical specifications (TS) surveillance requirements currently require verification of the actuation (closing) capability of each reactor instrumentation line EFCV every 18 months (or 24 months depending on the plant refueling schedule). This is typical for most BWR plants. The proposed change by the licensee revises the surveillance frequency by allowing a “representative sample” of EFCVs to be tested every 18 months. The “representative sample” is based on approximately 17 percent of the EFCVs being tested each refueling outage such that each valve is tested at least once every 10 years (nominal).

The reactor vessel instrument lines at VY include flow restricting orifices upstream of the EFCVs to limit reactor coolant flow in the event of an instrument line break. The licensee states that: (1) in the unlikely event where an EFCV fails to function properly concurrent with a postulated line break outside containment, orificing and small tube diameters limit flow rates, thus ensuring that the integrity and functional performance of secondary containment is maintained; and (2) the coolant loss under such a scenario is well within the makeup capability of reactor coolant supply systems, and the potential off-site radiological consequences have been evaluated to be substantially below the limits of 10 CFR Part 100.

The VY TS surveillance requires the EFCVs to be tested for proper operation in accordance with the IST program. The VY IST program has deferred the quarterly testing of these valves based on the provision of the Code that states "if exercising is not practical during plant operation or cold shutdowns, it may be limited to full-stroke testing during refueling outages." Based on the Code provision above, and the Refueling Outage Justification (ROJ) number ROJ-V01, EFCVs at VY are currently tested once every refueling outage (18 months).

The licensee's justification for the relief request is based on General Electric Nuclear Energy (GE) Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" dated June 2000. The topical report provided: (1) an estimate of steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close, and (2) an assessment of the radiological consequences of such a release. The staff reviewed the GE topical report and issued its evaluation on March 14, 2000. In its evaluation, the staff agreed that the test interval could be extended up to a maximum of 10 years. In conjunction with this finding, the staff noted that each licensee that adopts the relaxed test interval program for EFCVs must have a failure feedback mechanism and corrective action program to ensure EFCV performance continues to be bounded by the topical report results. Also, each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure analysis, and release frequency analysis to confirm that they are bounded by the generic analyses of the topical report.

In this safety evaluation, the staff has reviewed the licensee's proposal for applicability to GE Topical Report NEDO-32977-A and conformance with approved staff guidance regarding radiological dose assessment, EFCV failure rate and release frequency, and the proposed failure feedback mechanism and corrective action program. Based on its review, the staff concludes that the radiological consequences of an EFCV failure are sufficiently low and acceptable, and that the alternative testing in conjunction with the corrective action plan provides a high degree of valve reliability and operability. Additionally, an orifice is installed just inside the drywell on all but the jet pump instrument lines, which are small diameter tubes. The orifice and small tube diameter limit leakage to a level where the integrity and functional performance of secondary containment and associated safety systems are maintained, and the coolant loss is within the capability of the reactor coolant supply systems. Therefore, the staff finds that the licensee's proposed test alternative provides an acceptable level of quality and safety.

3.0 CONCLUSION

Based on the preceding evaluation, the staff finds the proposed relaxation of VY EFCV test frequency, which would allow a representative sample of EFCVs to be tested every 18 months with all EFCVs being tested at least once every 10 years (nominal), to be acceptable. Therefore, the licensee's proposed alternative to the Code testing requirements is authorized pursuant to 10 CFR 50.55a(a)(3)(i) based on the alternative providing an acceptable level of quality and safety.

Principal Contributor: Y. S. Huang

Date: April 2, 2001