VERMONT YANKEE
NUCLEAR POWER CORPORATION

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August 18, 1995
BVY 95-89

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

References: Operating License DPR-28
Docket No. 50-271
Reportable Occurrence No. LER 95-15

Dear Sir:

As defined by 10 CFR 50.73, we are reporting the attached Reportable Occurrence as LER 95-15.

Very truly yours,

Robert J. Wanczyk

Robert J. Wanczyk
Plant Manager

RJW/dm
cc: Regional Administrator
USNRC
Region I
475 Allendale Road
King of Prussia, PA 19406

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1/1

NRC Form 366 (5-92) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO. 3150-0104
EXPIRES 5/31/95
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
VERMONT YANKEE NUCLEAR POWER STATION	05000271	01 OF 03

TITLE (4) Technical Specification 4.7.A.4 Leakage Rate Exceeded Due to Leakage from the Inboard Flange of Valve AC-8

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NO.(S) 05000
04	25	95	95	-- 15 --	00	08	18	95	N/A	

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: CHECK ONE OR MORE (11)							
			20.402(b)		20.405(c)		50.73(a)(2)(iv)		73.71(b)
POWER LEVEL (10)	100		20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
			20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER:
			20.405(a)(1)(iii)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)(A)		(Specify in Abstract below and in Text, NRC Form 366A)
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)(B)		
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NO. (Include Area Code)
ROBERT J. WANCZYK, PLANT MANAGER	802-257-7711

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
N/A					N/A				
N/A					N/A				

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MO	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X NO			N/A	

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

During the stabilization phase of the Integrated Leakage Rate Test (ILRT) excessive leakage at penetration X-26 on the primary containment side flange of valve SB-16-19-8 (AC-8) was identified. The leakage was subsequently calculated to be 0.1316 wt%/day. Technical Specification Surveillance Requirement 4.7.A.4 specifies that the leakage from any one valve shall not exceed 5% of Ltm (0.0400 wt%/day). The leakage at the inboard flange of AC-8 was eliminated by tightening the flange bolts after a leakage assessment was completed prior to the testing phase of the ILRT. A root cause analysis determined that the local leakage rate surveillance test procedure did not provide the necessary steps to pressurize the primary containment side flange of valve AC-8 in a manner that meets the requirements of section III.C.1 of 10 CFR 50 Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors". A review of all containment penetrations with similar penetration configurations concluded that there were seven additional valves with similar configurations.

The as-found total Type B and Type C leakage rate remained below 0.60 L_a and as-found ILRT leakage remained below 0.75 L_a in accordance with 10 CFR 50 Appendix J sections III.C.3 and III.A.5.(b)(2) respectively. The as-left leakage rates for the primary containment were below the 10 CFR 50 Appendix J and Technical Specification limits.

DESCRIPTION OF EVENT

During a review (07/20/95) of the test data of the Integrated Leakage Rate Test (ILRT) performed during April of 1995, it was determined that Technical Specification Surveillance Requirement 4.7.A.4 was exceeded for valve SB-16-19-8 (AC-8). Technical Specification Surveillance Requirement 4.7.A.4 states that the leakage from any one isolation valve shall not exceed 5% Ltm (0.0400 wt%/day). Contrary to this requirement, leakage for AC-8 was determined to be 0.1316 wt%/day.

ANALYSIS OF EVENT

The primary containment is pressurized on a periodic basis to ensure its leak-tight integrity. During the stabilization phase of the April 1995 ILRT, a leak was discovered at the primary containment side of valve AC-8. This valve is an 18 in. diameter Allis Chalmers Model 150R butterfly valve on the drywell purge supply isolation line. This valve is bolted in place between two pipe flanges.

During the 1995 refueling outage, this valve was local leakage rate tested utilizing surveillance procedure OP 4030 "Type B and C Primary Containment Leak Rate Testing". The test method identified in OP 4030, applied test pressure in the reverse direction (non-accident direction). Section III.C.1 of Appendix J states that pressure shall be applied in the same direction as that when the valve would be required to perform its safety function, unless it can be determined that the results from the test for a pressure applied in a different direction will provide equivalent or more conservative results. Since the primary containment side flange of AC-8 was not exposed to test pressure during local leakage rate testing done per OP 4030, this test method did not provide equivalent or more conservative results.

A review of all containment penetrations concluded that there were seven additional valves with similar configurations. These valves did not experience noticeable leakage during the 1995 ILRT.

During this event, the total Type B and Type C leakage rate remained below 0.60 L_a and overall primary containment leakage remained below 0.75 L_a in accordance with 10 CFR 50 Appendix J sections III.C.3 and III.A.5.(b)(2) respectively. Therefore, at no time as a result of this condition was the health and safety of the public affected.

CAUSE OF EVENT

Root Cause: Inadequate Procedure

Surveillance procedure OP 4030 did not provide the necessary steps to pressurize the primary containment side flange of valve AC-8 in a manner that meets the requirements of section III.C.1 of 10 CFR 50 Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors".

Contributing Causes:

1. Inadequate resolution of a condition identified during an Operating Experience Review. Two similar events were identified in USNRC Information Notice (IN) 92-20, "Inadequate Local Leak Rate Testing" dated March 3, 1992. The conditions identified in IN 92-20 were determined to be applicable but the recommended actions were insufficient to test these valves in a manner that would challenge the primary containment side flange during periodic local leakage rate tests.

2. Untimely resolution of a vendor recommendation. A Vermont Yankee consultant completed a review to identify Vermont Yankee commitments and compliance with the requirements of 10 CFR 50 Appendix J. A report issued by the consultant dated May 5, 1994 recommended that these flanges be added to the local leakage rate test program

and tested on a periodic basis. Action had not been taken to incorporate the testing methods for this and the other affected valves.

CORRECTIVE ACTIONS

Immediate:

1. The flange for valve AC-8 was tightened and the leakage path was eliminated.

Short Term:

1. A review of all containment penetrations similar to X-26 was perform to determine if other containment isolation valves are susceptible to leakage past e primary containment side flange that are not local leakage rate tested. Seven additional valves in three containment penetrations were identified as being improperly local leakage rate tested. The leak-tight integrity of the additional valves was verified during the ILRT.

Long Term:

1. The local leakage rate surveillance test procedure (OP 4030) will be revised to test valve AC-8 and the seven additional valves in a manner that will challenge the inboard flange of the valve(s) on a periodic basis in accordance with the requirements of 10 CFR 50 Appendix J subsection III.C.1. This action is expected to be completed by 12/31/95.

2. Perform a re-review of all other relevant 10 CFR 50 Appendix J industry operating experience documents (NRC Information Notices, NRC Bulletins, NRC Generic Letters, INPO SOERs, INPO SERs and etc.) for applicability to Vermont Yankee. This action is expected to be completed by 03/31/96.

3. Perform a complete review of all Appendix J consultant recommendations in the report dated May 5, 1994 and to verify their applicability (if any) to Vermont Yankee. This action is expected to be completed by 09/30/95.

4. Perform a self-assessment of the Vermont Yankee Operating Experience Review Program to (1) ensure that industry documents are adequately reviewed and dispositioned and (2) that applicable vendor reports containing recommendations are dispositioned in a timely manner. These actions are expected to be completed by 12/31/95.

ADDITIONAL INFORMATION

No similar events regarding the failure to adequately local leak rate test the primary containment side flanges of butterfly valves have been reported to the Commission in the past five years.