VERMONT YANKEE NUCLEAR POWER CORPORATION

P.O. Box 157, Governor Hunt Road
Vernon, Vermont 05354-0157
(802) 257-7711

June 27, 2001
BVY 01-50

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)
Reportable Occurrence No. LER 2001-02, Rev. 0

As defined by 10CFR50.73, we are reporting the attached Reportable Occurrence as LER 2001-02, Rev. 0.

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Kevin H. Bronson
Plant Manager

cc: USNRC Region I Administrator
USNRC Resident Inspector - VYNPS
USNRC Project Manager - VYNPS
Vermont Department of Public Service

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-1998)

LICENSEE EVENT REPORT (LER)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000271	Page 1 of 3

TITLE (4)
HPCI TURBINE EXHAUST LINE CHECK VALVE TESTING RESULTS IN PRIMARY CONTAINMENT LEAK RATE IN EXCESS OF ALLOWABLE DUE TO VALVE DESIGN.

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 NUMBER
05	02	01	2001	02	00	06	27	01	N/A	

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR • : (Check one or more) (11)							
			20.2201(b)		20.2203(a)(2)(v)	X	50.73(a)(2)(i)		50.73(a)(2)(viii)
POWER LEVEL (10)	000		20.2203(a)(1)		20.2203(a)(3)(i)		50.73(a)(2)(ii)		50.73(a)(2)(x)
			20.2203(a)(2)(i)		20.2203(a)(3)(ii)		50.73(a)(2)(iii)		73.71
			20.2203(a)(2)(ii)		20.2203(a)(4)		50.73(a)(2)(iv)		OTHER
			20.2203(a)(2)(iii)		50.36(c)(1)	X	50.73(a)(2)(v)		Specify in Abstract below or in NRC Form 366A
			20.2203(a)(2)(iv)		50.36(c)(2)		50.73(a)(2)(vii)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Kevin H. Bronson, Plant Manager	(802) 257-7711

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (12)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	BJ	ISV	E334	YES		N/A				
N/A						N/A				

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

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

On 05/02/01, Vermont Yankee (VY) determined, by containment isolation valve leak rate testing of the High Pressure Coolant Injection (HPCI) steam turbine exhaust line nozzle check valves, that each of the redundant containment isolation nozzle check valves allowed leakage beyond the prescribed limit. VY primary containment isolation valves are tested in accordance with 10CFR50 Appendix J. The plant condition (refueling outage) at the time of the discovery did not require that primary containment integrity be maintained. VY made modifications and repairs to the nozzle check valves and satisfactorily tested them to restore primary containment integrity prior to restart of the plant. The cause of this event is concluded to be the design of the internal nozzle check valve components. Leakage through this pathway would be into the HPCI system piping, which is contained within secondary containment. Leakage out of the HPCI system piping would be treated by the Standby Gas Treatment System prior to discharge. The VY design basis LOCA predicts radioactive releases based upon conservative assumptions. The predicted doses are in the millirem range, are 3-4 orders of magnitude below 10CFR100 criteria, and any additional leakage through the HPCI steam turbine exhaust line nozzle check valves would not have resulted in a significant increase in risk to public health and safety.

Description of Event

On 05/02/01, with the plant shutdown for refueling, it was determined during 10CFR50, Appendix J, Type C, primary containment leak rate testing that the leakage rates of both (in series) HPCI (EIIS=BJ) steam turbine exhaust nozzle check valves exceeded the Technical Specification (TS) allowable leak rate limits. This event is reportable under 10CFR50.73(a)(2)(I)(B) as a condition prohibited by the plants TS and 10CFR50.73(a)(2)(v) as a condition that could have prevented the fulfillment of the safety function needed to mitigate the consequences of an accident.

The leakage of both HPCI steam turbine exhaust nozzle check valves, caused the combined local leak rate test (Type B and C tests) acceptance criterion of <0.60 L_a, calculated on a minimum pathway basis to be exceeded.

Background

In 1999, VY installed two new nozzle check valves in the HPCI steam turbine 20" exhaust line. The nozzle check valves are similar and utilize a disc floating on springs and radial guides to check flow and have flange connections on each end for ease of installation/removal. These valves replaced two swing-check valves that were original plant equipment. They were satisfactorily leak rate tested after installation and not tested again until this refueling outage.

Both valves were designed to meet the specifications of the application in the VY HPCI system. Those specifications took into account the need for HPCI to automatically start on demand and to permit operation under all design flow conditions while not creating excessive backpressure on the steam turbine. The valves are in series in the HPCI turbine exhaust line and the downstream valve will see a slightly lower operating pressure. Additionally one valve is physically located horizontally and the other vertically due to the existing piping configuration. The horizontal valve (V23-3) has stronger springs because the disc weight is being supported by the diffuser and radial guides and needs to be stronger to overcome the associated friction. The springs in the vertical valve (V23-4) are designed with a lesser spring force because the weight of the disc aids in closing/seating the disc.

Cause of Event

The cause of the failure is concluded to be the design/sizing of the valve springs. The springs were specified to provide minimal spring force, to keep the HPCI turbine backpressure as low as possible. A contributing cause was believed to be the deformation of the radial guides. The spring design may have contributed to the deformation of the radial guides.

Analysis of Event

The HPCI system consists of a steam turbine assembly driving a constant-flow pump. Steam for the HPCI turbine is provided from the reactor and exhaust steam from the turbine is discharged to the suppression pool. The HPCI system is provided to assure that the reactor core is adequately cooled in the event of a small break in the nuclear system and loss of coolant which does not result in rapid depressurization of the reactor vessel, simultaneous with a loss of normal auxiliary power. There is no indication that HPCI system performance was negatively impacted during the previous operating cycle by any problem(s) associated with these nozzle check valves. Periodic HPCI system surveillance testing was satisfactorily performed during this period.

The primary containment consists of a drywell, which encloses the reactor vessel and recirculation system, a pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and the suppression chamber, isolation valves and other associated components. The primary containment is capable of withstanding peak pressure which is postulated from a loss of coolant accident, to limit

the release of fission products to the plant environs, so that off-site doses would be well below limits.

The leakage from the primary containment would be from the Torus air space via the HPCI vacuum breaker system (2" piping and two 3" in-series nozzle check valves) into the HPCI turbine steam exhaust system (20" piping and two in-series nozzle check valves) and exceeded the leakage assumed in the current analysis. The leakage would be to the HPCI turbine, where it can be presumed to leak into the HPCI room (enclosed within secondary containment) through the turbine shaft seals. Any leakage into secondary containment through the turbine shaft seals would be captured, filtered and processed through the Standby Gas Treatment System (SBGT), thus limiting any potential off-site dose consequences.

The design basis LOCA for VY, as detailed in Section 14.6 of the FSAR, predicts the release of noble gases and iodine based upon conservatively estimated fuel failure and fuel gap inventories. Credit was taken for iodine removal in the drywell and torus, and by the standby gas treatment system. Conservative meteorology was used to determine off-site doses at a location where the terrain height is equal to the height of the primary vent stack. The resulting doses were in the millirem range for both whole body and thyroid. These doses are 3-4 orders of magnitude below the 10CFR100 criteria and any additional leakage through the HPCI turbine steam exhaust valves would not have resulted in an unacceptable risk to the public.

It has been determined that this condition would not adversely impact the plant's containment Level-2 Risk Assessment. Specifically, this failure of primary containment integrity would not contribute to the determination of the Large Early Release Frequency (LERF) due to the small size of the air-space release flow path (≤ 2" diameter piping) and the fact any release via the main exhaust line discharge would be "scrubbed" by the suppression pool, reducing the postulated release magnitude by two release categories (i.e., from High to Low).

Corrective Actions

1. The valve springs were replaced with stronger springs. The HPCI system has been full flow tested and turbine backpressure is acceptable with the stronger springs. This action is complete.

2. The radial guides were replaced with stronger and thicker Inconel radial guides. This should preclude the potential for buckling or deformation of the radial guides. This action is complete.

3. The seat and disc for V23-3 was lapped to repair some minor seat damage. This action is complete.

4. Both valves were satisfactorily leak rate tested after repairs/modifications were complete.

5. Follow-up verification will consist of performing leak rate testing of these valves at the next scheduled refueling outage, in accordance with 10CFR50, Appendix J.

Additional Information

In the past 5 years, similar events have been reported as:

LER 98-009 "Main Steam Isolation Valve Leakage Exceeds Technical Specification Limit which could have Impacted the Ability of a System to Mitigate the Consequences of an Accident"

LER 98-025 "Scram Discharge Volume Valve Closing Time Excessive due to Undersized Actuators"