Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

May 26, 2004
BVY 04-051

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 2004-001-00

As defined by 10CFR50.73, we are reporting the attached Reportable Occurrence LER 2004-001-00.

Sincerely,

ENTERGY NUCLEAR OPERATIONS, INC.
VERMONT YANKEE

Michael Desilets for KHB GMPO
Michael P. Desilets
Technical Support Manager

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

JE22

NRC FORM 366 (7-2001)	U.S NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Main Steam Isolation Valve Leakage Exceeds a Technical Specification Leakage Rate Limit

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000 -
04	05	2004	2004 -	001	- 00	05	26	2004	FACILITY NAME N/A	DOCKET NUMBER 05000 -

9. OPERATING MODE: N

10. POWER LEVEL: 0

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2201(d)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(1)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(iii)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	
☐ 20.2203(a)(2)(iv)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	
☐ 20.2203(a)(2)(v)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(viii)(A)	
☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(B)	

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Mike Desilets, Technical Support Manager	(802) 257-7711

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

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
D	SB	ISV	E331	Yes	E	SB	ISV	E331	Yes

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE)	☒ NO		N/A	N/A	N/A

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

On 04/05/04, with the plant shutdown for refueling, the "B" inboard main steam line isolation valve exceeded its allowable leakage rate limit during local leakage rate testing. Technical Specification (TS) 3.7.A.4 states that whenever primary containment is required, the leakage from any one main steam isolation valve (MSIV) shall not exceed 31 standard cubic feet per hour (scfh) at 44 psig. The leakage rate test result on the "B" main steam line outboard MSIV was completed satisfactorily. Additionally, the total combined leakage rate from all Primary Containment penetrations was less than the maximum allowable leakage rate limit of 0.8% air weight per day as specified in TS 6.7.C, thereby remaining within the design bases of the plant. The cause of the leakage rate test failure was due to a binding of the stem and packing follower from scoring and galling that was compounded by each subsequent cycling of the valve during scheduled surveillance activities. This mechanical degradation occurred as a result of maintenance instructions provided during prior maintenance activities that lacked specificity regarding tolerances.

The MSIV was disassembled, repaired, modified, cleaned, tested satisfactorily and returned to service. This condition was bounded by the primary containment design basis analysis and therefore, did not significantly degrade plant safety. There was no increased risk to the health and safety of the public as a result of this event.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

DESCRIPTION:

On 04/05/04, with the plant shutdown for refueling, it was determined that the leakage rate of the inboard Main Steam (EIIS=SB) Isolation Valve (EIIS=ISV) in the "B" main steam line exceeded its Technical Specification (TS) allowable leakage rate limit. This event is reportable under 10CFR50.73(a)(2)(i)(B) as a Condition Prohibited by Technical Specifications. TS 3.7.A.4 states that whenever primary containment is required, the leakage from any one isolation valve shall not exceed 31 scfh at 44 psig (Pa). Contrary to this requirement, the inboard valve on the "B" main steam line exceeded this value with a leakage rate of 219.86 scfh. Although the inboard "B" MSIV exceeded the allowed 31 scfh leakage rate, the outboard isolation valve on the "B" main steam line was also tested and found to be within the TS limits for surveillance testing with a leakage rate of 0.68 scfh. Also, the total primary containment combined leakage rate was less than the maximum allowable primary containment leakage rate limit of 0.8% air weight per day as specified in TS 6.7.C and therefore, was within the design bases of the plant. Immediate compensatory actions were not required as the plant was shutdown when this condition was discovered.

ANALYSIS:

There are four main steam lines, each having one inboard and one outboard isolation valve. The "B" main steam line inboard MSIV exceeded the TS requirement with leakage rate test result of 162.38 scfh at a test pressure of 24 psig that correlates to a leakage rate of 219.86 scfh at the design bases accident pressure of 44 psig. The outboard MSIV passed the leakage rate test at 0.50 scfh at 24 psig (0.68 scfh at 44 psig).

The safety objectives of the MSIVs are to limit the release of radioactive material by closing the nuclear system process barrier and the primary containment barrier, and to limit the loss of reactor cooling water if a major steam leakage occurred from the main steam system outside of the primary containment

For design bases analysis of primary containment, a maximum allowable accident leakage rate of 1.5 wt%/day (600.39 scfh) at 44 psig for a loss-of-coolant accident (LOCA) results in acceptable doses pursuant to the limits provided in 10CFR100. For conservatism, this value is reduced by a factor of two. A value of 0.8 wt%/day (320.21 scfh) at 44 psig is the established limit for the maximum allowable (test) leakage rate (La); per TS 6.7.C. Additional conservatism is also added to the MSIV leakage rate testing.

A leakage rate test failure that is less than or equal to 1.5%wt/day, would demonstrate that primary containment was able to properly complete its required safety function. Although the leakage rate was above the TS limit for a single MSIV, the total primary containment combined leakage rate on a minimum pathway basis was 76.12 scfh. This value is below the TS limit for the combined local leakage rate test (type B and C test) acceptance criterion of less than 0.6 La (192.13 scfh) and below the conservative analyzed limit for primary containment of 0.8 %wt./day (320.21 scfh) by a factor of 4.

When considering post-LOCA Control Room habitability, the initial conditions used in the radiological analyses utilized a main steam line total combined leakage rate of 62 scfh with no credit taken for pressure

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

reduction (leakage rates maintained at a maximum value for thirty days). The total measured leakage rate on a minimum pathway basis from the main steam lines was 23.59 scfh, corrected to 44 psig. This consequently demonstrates that Control Room habitability was not adversely affected.

As a result of the preceding analysis, no increase to off-site or Control Room dose would occur as a result of this condition during a design basis loss of coolant accident. The primary containment total combined leakage rate was within the primary containment design basis analysis and did not significantly degrade plant safety or pose an increased risk to plant personnel or the general public.

Additionally, radioactive material associated with MSIV leakage would hold up and eventually pass through the turbine and condenser where most of the iodine would be removed by deposition, impaction and adsorption. Subsequent leakage from the condenser would be released via the turbine building ventilation system to the primary vent stack.

CAUSE:

The cause of the leakage rate test failure was due to a binding of the stem and packing follower from scoring and galling that was compounded by each subsequent cycling of the valve during scheduled surveillance activities. This mechanical degradation occurred as a result of maintenance instructions provided during prior maintenance activities that lacked specificity regarding tolerances.

Prior to 1999, the "B" MSIV stuffing box was mechanically cleaned without adequate control of the work activity. The significance of the tolerance in this aspect of reassembly was not recognized, therefore, the packing/stuffing box clearance was not verified to be within the design allowable clearance before assembly. The slightly oversized condition of the stuffing box allowed the packing spacer to contact the stem, leading to stem scoring and galling that ultimately resulted in the stem and packing follower binding in the seated position to a point where the valve failed the post-shutdown leakage rate test. Additionally, the design change process utilized in 1989 that was in place when the packing configuration was changed, failed to incorporate guidance from the vendor that the stuffing box dimensional clearances should be verified during assembly.

CORRECTIVE ACTIONS:

Immediate:

1. Initiated a Condition Report for the LLRT failure, and a Condition Report to specifically address the stem and packing follower galling issue.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Prior to Reactor Start Up:

1. Repaired, cleaned and modified the "B" MSIV. The stuffing box configuration was modified to utilize a custom fit carbon packing spacer in lieu of the cadmium plated carbon steel standard packing spacer and increased clearances in the backseat and packing follower areas.
2. Initiated and completed the planned contingency diagnostic inspection, repair, and refurbish plan for other valves that had indications of stem scoring.
3. Notified the nuclear industry of this condition, analysis, cause and corrective actions implemented to prevent recurrence through the INPO Nuclear Network (OE18258).

Long Term:

1. Review and update MSIV drawings to specifically identify critical design dimensions.
2. Develop a controlled MSIV maintenance procedure that adequately describes the required actions to be taken during disassembly and assembly of these valves.
3. Determine if additional essential valves are similarily affected and initiate corrective actions if necessary.
4. A Work Order Request has been inititated to install custom fit carbon packing spacers in the remaining MSIVs that were not disassembled during RFO-24, at the next opportunity that requires repacking.

ASSESSMENT OF SAFETY CONSEQUENCES:

During the last operating cycle, no events at the plant resulted in conditions that required the Main Steam Isolation Valves to be closed to prevent a release of radioactive materials. Since the primary containment total combined leakage rate was within TS limits, the MSIV leakage would not have resulted in an increase in off-site dose or inpacted Control Room habitability beyond analyzed limits, had a Design Basis Loss of Coolant Accident occurred. Therefore, this event did not significantly increase the risk to the health and safety of the public.

ADDITIONAL INFORMATION:

No similar events with a related cause have occurred within the past five years at Vermont Yankee.