Sienel, Beth

From:	Hamer, Mike
Sent:	Tuesday, May 25, 2004 3:05 PM
To:	Devincentis, Jim; Rogers, James; Ball, Mike; Underkoffler, Ted; Hockenberry, Dale; Pelton, David; Sienel, Beth; Wierzbowski, George
Subject:	PRO for CR-VTY-01017 - V23-3 failed LLRT

This event has been determined to be not reportable. This report does not alleviate the vendor from the requirements of § 21.21(c), that states;

"A dedicating entity is responsible for – (1) Identifying and evaluating deviations (deviations pertains to the procurement documents) and reporting defects and failures to comply associated with substantial safety hazards for dedicated items."

Please note the reporting difference between the dedicating entity and anyone else. VY reports conditions that **could have created a substantial safety hazard**, and the vendor (or dedicating entity) must report those conditions that are **associated with substantial safety hazards.** This difference in language within the regulations sets the vendor's threshold for reporting much lower than ours, since they are not always able to determine if a certain condition presents a substantial safety hazard at a particular station.

PRO-0401017 -
HPCI V23-3 fails...

A-69

INTEROFFICE MEMORANDUM
LICENSING
POTENTIALLY REPORTABLE OCCURRENCE REPORT

TO: MIKE DESILETS, TECHNICAL SUPPORT MANAGER

FROM: MIKE HAMER, TECHNICAL SPECIALIST III

SUBJECT: CR-VTY-2004-01017; V23-3 FAILED APPENDIX "J" LOCAL LEAK RATE TESTING

DATE: MAY 25, 2004

PRO NUMBER: PRO-0401017

EVENT DESCRIPTION:

On 04/10/04, with the reactor shutdown for a refueling outage, the HPCI Turbine Inboard Steam Exhaust Line check valve (V23-3) would not pressurize to 44 psig during LLRT. V23-3 was declared inoperable for primary containment integrity and Work Order Request 04-60967 was written for repair. V23-4 passed LLRT with essentially zero leakage (0.02 scfh).

V23-3 was removed from the system and bench tested prior to disassembly. The use of a shorter hose enabled the testers to achieve sufficient flow across the valve as necessary to create a differential pressure that caused the valve to close, seal and pressurize to 44 psig.

The following regulations were considered when determining reportability of this event.

<u>Operation or Condition Prohibited by Technical Specifications</u>

§50.73(a)(2)(i)(B) "[The licensee shall report:] Any operation or condition which was prohibited by the plant's Technical Specifications except when: (exceptions do not apply)

<u>Degraded or Unanalyzed Condition</u>

§50.73(a)(2)(ii) "[The licensee shall report] Any event or condition that resulted in:

(A) The condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or

(B) The nuclear power plant being in an unanalyzed condition that significantly degraded plant safety."

Event or Condition That Could Have Prevented the Fulfillment of a Safety Function

§50.73(a)(2)(v) "[The licensee shall report:] Any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shutdown the reactor and maintain it in a safe shutdown condition;
(B) Remove Residual Heat
(C) Control the release of radioactive material; or
(D) Mitigate the consequences of an accident."

DISCUSSION/BASES:

Background:

Both valves are 20" ANSI Class 15 lb. Nozzle Check Valves manufactured by Enertech. The valves have a large circumferential disk employing two concentric seating surfaces that mate to seating surfaces on the valve body. Concentric alignment of the disk to the valve centerline is maintained by use of radial guides attached to a fixed point on the diffuser and at the other end to guiding shoes that transmit the spring force to the disk.

A root cause analysis was performed to determine the cause of the LLRT failure of V23-3. The LLRT for this valve indicated that the valve did not seal at the required test pressure of 44 psig, therefore the actual leakage rate with the check valve while still installed in the system was indeterminate. V23-3 was declared inoperable. A leakage rate test was performed on the other check valve in the HPCI exhaust line (V23-4). This valve passed the leakage rate testing at 0.02 scfh, significantly below the test acceptance criteria of 15.5 scfh for each valve.

V23-3 was disassembled and all internal parts were examined for damage or for non-conformance with the valve design.

- The results of the examination indicated that the installed radial guides were not of sufficient strength (distortion and bending was evident) to absorb the load placed on the valve during HPCI turbine operation. The radial guide material was determined to have a yield strength approximately 60% below that required by the valve design. This lack of strength was attributed to improper heat treatment of the material during fabrication.
- A large variation in the spring rate between the three installed springs was observed and subsequently determined to have contributed to the failure of the radial guides.
- V23-3 is installed horizontally, and V23-4 is installed vertically in the steam exhaust line. This places more reliance on V23-3's radial guide and spring function to ensure the valve is fully closed. The additional force experienced by the V23-3 radial guides that contributed greatly to their distortion, is negligible with V23-4 solely due to their respective differences in configuration within the system.
- The HPCI system is run on a quarterly basis.

Therefore, it may be deduced that the LLRT failure was due to a combination of insufficient materials, the large variation in the spring rate between the three springs, the configuration of the valve within the system combined with the quarterly operation of the system.

New thicker guides of a modified design, intended to minimize the stress imparted during turbine operation, and new springs with a tighter tolerance were installed to alleviate these potential failure mechanisms (WOSE 2004-038). The valve passed the subsequent post-maintenance LLRT. Additional work has been performed IAW the WOSE to machine guiding shoes and washer plates to the manufacturer's (Enertech) specifications.

Reportability Considerations

For determining reportability for this type of test failure, it should be noted that the check valve did not fail in a manner that would prevent the HPCI system from injecting water into the reactor, or returning the exhaust steam to primary containment (Torus) when called upon to do so. This LLRT failure presents a challenge to maintaining primary containment integrity while the plant is operating.

Operation or Condition Prohibited by Technical Specifications

NUREG 1022, Rev. 2 states:

> "For testing that is conducted within the required time it should be assumed that the discrepancy occurred at the time of discovery unless there is firm evidence, based upon a review of relevant information such as the equipment history and the cause of failure, to indicate that the discrepancy existed previously."

Investigation results determined the condition that caused the leakage rate test failure occurred primarily from an inadequate dedication process used by the vendor for the radial guides and the spring rate. The deformation of the radial guides and the disparity in the spring rate ultimately resulted in V23-3 failing to adequately pressurize to 44 psig (Pa) as necessary to measure the leakage rate through the valve during as-found LLRT. Therefore, there is reasonable assurance that the LLRT failure was due to a combination of insufficient materials, the large variation in the spring rate between the three springs, the configuration of the valve within the system combined with the quarterly operation of the system. Since no operation of the system occurred immediately before the LLRT, there is firm evidence that this condition most likely existed since the last HPCI system quarterly operation surveillance. This is construed from the information presented as a "cause of failure" pertaining to the design and fabrication that this particular component (the check valve, not the system itself) was not capable of performing its design safety function for primary containment isolation in the event of an accident.

For a "Condition Prohibited by Technical Specifications" to be reportable, the condition must have existed for a time longer than the allowed LCO period for maintaining primary containment.

TS 3.7.D.2. provide direction for the Primary Containment LCO for Primary Containment Isolation Valves by stating the following:

> "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."

V23-4 was operable through the entire operating cycle. Therefore, the LCO was satisfied and the condition of V23-3 is not reportable in regard to this criterion.

Degraded or Unanalyzed Condition

NUREG 1022, Rev. 2 provides examples of reportable conditions. Example (5) that pertains to containment leak rate tests states;

> "Loss of containment function or integrity, including containment Leak rate tests where the total containment as-found, minimum-pathway leak rate exceeds the limiting condition for operation (LCO) in the facility's TS."
>
> "....Minimum-pathway leak rate means the minimum leak rate that can be attributed to a penetration leakage path; for example, the smaller of either the inboard or outboard valve's individual leak rates."

V23-4 in the same line had a test leak rate of 0.02 scfh, at 44 psig (Pa), well below the limit for each valve of 15.5 scfh. Consequently, this condition did not seriously degrade the nuclear power plant or its principal safety barriers and did not place the plant in an unanalyzed condition that significantly degraded plant safety.

Event or Condition That Could Have Prevented the Fulfillment of a Safety Function

As stated in 10 CFR;

> §50.73(a)(2)(vi) "Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery. However, individual component failures need not be reported pursuant to paragraph (a)(2)(v) of this section if redundant equipment in the same system was operable and available to perform the required safety function of design, analysis, fabrication, construction, and/or procedural inadequacies."

In compliance with the rule stated above, V23-4 remained operable as a redundant valve within the same system; for that reason alone, this event is not reportable as a safety system functional failure.

Part 21 Evaluation

A separate Part 21 evaluation is not necessary. These valves were installed in the plant when this condition was discovered. The preceding evaluations for Part 50 satisfy the regulatory responsibility for the evaluation of potential defects (see §21.2 below). Due to the configuration of the system, this condition could not have created a "Substantial Safety Hazard". V23-4 was capable, and remains capable of performing both of its design safety functions.

> §21.2 Scope
>
> (c) For persons licensed to operate a nuclear power plant under part 50 of this chapter, evaluation of potential defects and appropriate reporting of defects under §§ 50.72, 50.73 or § 73.71 of this chapter satisfies each person's evaluation, notification, and reporting obligation to report defects under this part and the responsibility of individual directors and responsible officers of such licensees to report defects under section 206 of the Energy Reorganization Act of 1974.

CONCLUSION: This event is **not reportable** as an LER pursuant to §50.73(a)(2)(i)(B), §50.73(a)(2)(ii) or §50.73(a)(2)(v).

RECOMMENDED: _______________, 5-25-04
Michael J. Hamer — Date
Technical Specialist III

APPROVED: _______________, 5/25/04
Michael P. Desilets — Date
Technical Support Manager

REFERENCES:

1) CR-VTY-2004-01017, Vermont Yankee Root Cause Analysis Report, Fluid Systems Engineering, 4/30/04.

2) Enertech Job #950010, Root Cause Analysis for Enertech 20 Inch –ANSI 150 Type DRV-B Nozzle Check Valve, Drawing Number MA21137, 4/26/04.

3) Technical Specification 3.7.D., Amendment 210, Pg. 158.

4) CR-VTY-2004-01017, Operability Recommendation, 4/10/04.