ENCLOSURE

PRIORITIZATION EVALUATION

Generic Issue No. 85

"Reliability Of Vacuum Breakers Connected To Steam Discharge Lines Inside BWR Containments"

ISSUE 85: RELIABILITY OF VACUUM BREAKERS CONNECTED TO STEAM DISCHARGE LINES INSIDE BWR CONTAINMENTS

DESCRIPTION

Historical Background

In Boiling Water Reactors (BWRs) safety relief valves (SRVs) are mounted on the main steam line inside the drywell. Each SRV discharge is piped through its own discharge line (tailpipe) to a point below the minimum water level in the primary containment suppression pool. A vacuum breaker (VB) valve is installed on each discharge line to admit drywell air into the discharge line after SRV actuation and closure. This prevents a vacuum from forming due to the condensation of leftover steam in the discharge line. Water in the suppression pool then will not be drawn up into the line. The VB valve is similar to a swing check valve with a disk that swings on a hinge pin to open, and a spring to return the disc to a closed position.

VB valves are also mounted on other steam lines, (the HPCI & RCIC turbine exhaust lines) that also discharge into the primary containment pressure suppression pool. However, the VB valves on these lines admit air from above the suppression pool (i.e., wetwell airspace) rather than drywell air.

Review of recent Licensee Event Reports (LERs) has shown several instances in which VB valves from various vendors and in different plants have failed to properly operate, indicating a potential generic problem. Based on the information provided for this safety issue[A], it appears that the cyclic impact of the disk on the VB valve seat due to steam discharge and condensing during SRV actuation or leakage presents a loading condition that has not been addressed in approved VB valve design and qualification requirements.

Safety Significance

Failure of a SRV discharge line vacuum breaker valve in the open position when combined with the failure of its SRV in the open position would result in an extended steam release to the drywell and would present the control room operators with a confusing set of indications, i.e. simultaneous indications of a stuck open or leaking SRV and a LOCA in the drywell. Several events have been reported in which the above scenario was evidenced. The most notable was the event at Hatch Unit 2 on August 25, 1982.[B] Misinterpretation and confusion on the control room operators part would be expected to increase the probability of operator error in the course of response to the event and might result in an increased likelihood of the event escalating into a severe core damage accident. Steam release to the drywell may also affect safety-related valves and instruments as well as increase the drywell temperature and pressure.

Failure of a SRV discharge line vacuum breaker valve in the closed or near closed position when combined with a second actuation of its SRV could result in hydrodynamic loads on the discharge line pipe and to the suppression chamber in excess of the design loads or could cause water hammer damage to the SRV. If the hydrodynamic loads are severe enough failure of the SRV discharge line or the suppression pool could occur with some probability that these failures would lead to a severe core damage event. Damage of the SRV could also lead to a greater frequency of challenges to the Reactor Protection system and the containment. Evaluation of the available LERs has not revealed a closed or near closed VB failure. Since a possible outcome of closed or near closed failure of a SRV VB valve would be to bypass the containment

pressure suppression pool and/or loss of containment integrity, this failure mode is evaluated in Generic Issue 61, "SRV Discharge Line Break in the BWR Wetwell Airspace of Mark I & II Containments."

Failure of HPCI or RCIC turbine exhaust line VBs in the closed or near closed position when combined with multiple actuations of the turbine driven pump could also result in excessive hydrodynamic loads on the turbine exhaust line or the containment wet well structure or water hammer damage to the HPIC or RCIC Turbine System. Failure of HPCI or RCIC turbine exhaust line VBs in the open or leaking position could result in a suppression pool bypass leak to the wetwell air space and present the possibility of excessive containment pressurization. Since both the open and closed failure modes of the HPCI/RCIC turbine exhaust line VBs could result in bypass of the containment pressure suppression pool and/or loss of containment integrity, failure of HPCI and RCIC turbine exhaust line VBs is also evaluated in GI 61.

As a result, Generic Issue 85 is limited to only the effects of SRV VB leakage failures.

Possible Solution

The reliability of VBs connected to SRV discharge lines inside BWR containment is assumed to be improved by development of NRC approved design criteria for VB valves, modification of current valve design(s) and a prototype qualification testing program(s). Licensees are assumed to replace the existing VB valves with a valve of the new qualified design. In addition, a Technical Specification is assumed that would require periodic operability testing of the VB valves.

All BWR plants using VBs connected to SRV discharge lines inside containment would be affected by this issue. The resolution of this issue is applicable to all BWRs. Therefore, this issue is applied to 44 BWRs with an average remaining lifetime of 27.4 years.

PRIORITY DETERMINATION

The prioritization of this issue is based in part on analysis performed by PNL[C] and in part on analysis by the NRC staff. Since the issue in question is generic and applies only to BWR plants, the Grand Gulf probabilistic risk assessment was utilized as the basis for estimating the risk reduction which might be achieved by the assumed resolution of the issue.

Frequency/Consequence Estimate

For the case in which the SRV discharge line vacuum breaker valve is assumed to fail open (i.e., the valve will not reseat under SRV discharge flow or the disc is lost), significant pressurization and steam accumulation in the drywell will only occur for a prolonged SRV discharge. Thus, this effect will only be seen for those events for which the SRV does not properly reseat. Of the Grand Gulf dominant risk sequences, only the T.P.Q.I and T.P.Q.E sequences are affected by the failure of an SRV to reseat (i.e., event P). T is the frequency of a reactor transient (7.2/ry), P is the probability of SRV failure to reset (0.1/demand), Q is the probability of failure of the power conversion system (1.0/demand), I is the probability of failure of the residual heat removal system to remove decay heat from the suppression pool within 28 hours (8.0×10^{-5}/demand) and E is the probability of failure of emergency core cooling (1.2×10^{-5}/demand). Prolonged SRV release to the drywell through a failed open VB is assumed to result in a severe core damage event, through

control room operator errors. Exposure of safety related instruments and equipment in the drywell was not considered because the environmental design loads for safety related equipment in the drywell exceed the expected environment following this event. Since containment failure due to H_2 Burn (γ) or steam explosion (α) is independent of the location of the SRV steam discharge to the containment, the T.P.Q.I.α and T.P.Q.E.γ sequences were eliminated as affected cut sets. This leaves only the T.P.Q.I.δ and T.P.Q.E.δ scenarios where δ represents the probability of long-term containment failure due to steam or non-condensable vapor overpressure. Examining these scenarios reveals that in the individual cutsets the only events requiring control room operator actions are the RECOVERY probability for the T.P.Q.I.δ sequence and the OP probability for the T.P.Q.E.δ sequence. In the Grand Gulf PRA, RECOVERY is defined as failure to restore maintenance/test faults or to take other corrective actions within 28 hours and is a subelement of event I. Since confusion over whether a LOCA had occurred or whether a SRV and VB were stuck open or leaking is assumed to hamper only the initial diagnosis of the event and very early response actions, we assumed the probability of RECOVERY would be unchanged by an open failure of the VB; therefore, the T.P.Q.I.δ sequence was also eliminated as an affected dominant risk sequence.

In the T.P.Q.E.δ sequence, OP is defined as the failure of the operator to manually initiate the automatic depressurization system (ADS)and is a subelement of event E. Since the OP event is a short-term operator response, we assumed control room confusion resulting from failure of the VB in the open position combined with a stuck open SRV would increase the likelihood of the OP event by an order of magnitude from .0015 to .015/demand.

Analysis of the VB failure data, provided in the request for classification of this concern as a generic issue by PNL, resulted in a calculated VB failure rate (X) of 0.0093/demand. Since this issue involves a failed open VB in conjunction with a malfunctioning SRV and the other failures, the affected dominant sequence becomes T.P.X.Q.E.δ.

The Grand Gulf PRA did not assume VB failure in the development of the event trees resulting in severe core damage. Therefore, we calculated a modified base case frequency for the T.P.X.Q.E.δ scenario of 2.43×10^{-8}/RY. Assuming that resolution of the issue (i.e., improving VB reliability) will result in an order of magnitude reduction in the VB failure rate (i.e., $\dot{X}^* = 0.00093$/demand), the post implementation frequency of the T.P.X*.Q.E.δ scenario is 2.43×10^{-9}/RY. Therefore, resolution of this issue would result in a reduction in the frequency of offsite release from a damage event of 2.18x10-8/RY and a reduction in core-melt frequency of 4.36×10^{-8}/RY.

The T.P.Q.E.δ event results in a BWR Category 4 offsite consequence (6.1×10^{5} man rem/event). Multiplying the reduction in dominant risk event frequency (2.18×10^{-8}/RY) by the Category 4 consequence (6.1×10^{5} MR/event), the number of applicable plants (44) and their average remaining lifetime (27.4 years) results in a public risk reduction of 16.0 Man Rem. This is the reduction in public risk and frequency of core-melt which might be provided by the improvement of SRV VB reliability for the fail open mode of VB failure.

Cost Estimate

The solution to this generic issue is assumed to result in a generic program for the development of a staff approved design criteria for the SRV VB valves, modifications of the valve design and a qualification program. The existing valves would be replaced and a periodic operability test would be required.

NRC Costs

It was assumed that cost for the development of design criteria and establishing an operability test for VBs would be sponsored by the NRC. It is assumed that a major portion of dynamic model development and engineering data is already available for use in establishing an adequate VB design

criteria. The NRC effort was estimated to require 1 man year for the development of the resolution of this issue for a cost of $100,000.

NRC review of the industry implementation of the resolution for the issue, i.e., selection and installation of a new VB valve and Technical Specifications for the operability testing of the valve, are assumed to require 1 man week of staff effort per plant at a cost of $2,270/man-week or $100,000 total cost.

About 0.1 man-wk/plant year is estimated for NRC review of periodic operability tests for VB valves. At $2,270/man-wk, this results in an estimated cost of about $275,000 over the remaining lifetime of the 44 BWR plants.

The total NRC costs estimated for the resolution and implementation of the issue is thus calculated to be about $475,000.

Industry Cost

The estimated costs for industry implementation of the solution to this issue (i.e., replace the valves and test them periodically) is estimated by PNL, based on conversations with Browns Ferry staff involved in their recent testing and maintenance effort on the VB valves at their plant.

PNL has estimated the purchase of new VB valves and initial operability tests to total about $54,000/plant, for a cost of about $2,400,000 to the industry. We have added to that cost an additional $500,000 which we estimate the industry would be required to spend to fund a VB valve generic qualification program. This results in a total industry estimated cost for implementation of this issue of $2,900,000.

The operability test requirement will require an estimated 0.5 man day/plant year of industry effort over the remaining lifetime of the BWR plants, or a total testing cost of about $275,000.

Thus, total industry cost is estimated to be $3,175,000 for resolution of this issue.

Total Cost

The total cost for resolution and implementation of this issue, i.e., NRC cost plus industry cost is thus estimated to be $3,650,000.

VALUE/IMPACT ASSESSMENT

Based on a total public risk reduction of 16 man-rem and a total cost of $3.65 million, the value/impact score is given by:

$$S = \frac{16 \text{ man-rem}}{\$3.65\text{M}}$$

$$= 4.4 \ \frac{\text{man-rem}}{\$\text{M}}$$

Other Considerations

Replacement and operability testing of the SRV VB valves will require that the plant operators perform almost all of the work in the drywell in close proximity to the SRV valves. Using the same man hours estimated for valve replacement and testing for only the operating BWR plants and an assumed 35 millirem/hr field in the drywell, a total operational exposure of about 200 man-rem is estimated.

We assumed that in every instance in which an SRV leaks or fails to close and its associated VB valve leaks to the drywell, a plant shutdown is required and the source of drywell pressurization and temperature increase must be found and corrected. Using the previous estimates of the frequency of VB failure and SRV leakage, an estimated two-day shutdown for each VB failure and a replacement power cost of $300,000/day, we have estimated that over the lifetime of the 44 BWR plants, these events will cost the industry about $40 million in lost power production. When discounted to present worth, this is equivalent to about $20 to $25 million.

When a SRV vacuum breaker valve (VB) failure is detected the valve is repaired or replaced. Resolution of this issue, i.e., placement of all existing SRV VBs with valves with an improved reliability, would therefore reduce the number of VB failures and accordingly reduce the number of VB replacements over the life of the BWR plants. Using the estimated current failure rate, the frequency of challenges (transients) and an average number of SRV VBs per plant (9), we estimate about 725 expected SRV VB failures over the remaining lifetime of the BWR plants. Improving the reliability of the VB by an order of magnitude would be expected to eliminate about 650 VB failures and, therefore, reduce the number of VB replacements by 650. Assuming 2 man-days labor to replace a failed VB and a 35 millirem work environment, we estimate that resolution of this issue would save about 350 man-rem of occupational exposure.

The averted occupational exposure, due to reduction in core melt frequency, is estimated to be less than 1 man-rem. Therefore, when all components of occupational exposure are considered, it appears that resolution of this issue would result in a net savings of about 150 man-rem of occupational exposure.

CONCLUSION

The calculated potential public risk reduction and core melt frequency reduction which might be obtained by the resolution and implementation of Generic Issue 85 are so small as to not justify further efforts on the issue. Therefore, we recommend that this issue be DROPPED from further consideration. However, it should be noted that although from the risk based regulatory perspective we recommend dropping the issue, there appears to be a large economic incentive to the industry to improve the reliability of SRV vacuum breaker valves.

References:

A. Memorandum for W. Minners from B. Siegel, "Proposed Generic Issue - Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments," November 3, 1983.

B. "Preliminary Case Study for the Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982," NRC Office for Analysis and Evaluation of Operational Data, Stuart D. Rubin, August 1983.

C. NUREG/CR-2800 Supplement, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development."