

# Valve and Piping System Test Results and Effects on Regulatory Standards

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## INTRODUCTION

The Idaho National Engineering Laboratory (INEL), under the sponsorship of the U.S. Nuclear Regulatory Commission (USNRC), is performing research to provide a technical basis for the resolution of specific generic issues and severe accident concerns and to provide information that will help to develop and improve mechanical equipment qualification standards. Research results are being used for improving plant safety at design basis loadings and for quantifying safety margins for severe accident loadings.

This paper summarizes the objectives, the research, selected results, and regulatory applications for the following four INEL research tasks:

- Nuclear Containment Purge and Vent Valves Closing in Design Basis Loss-of-Coolant Accident (LOCA) Environment (TMI Action Plan, Item II.E.4.2), reported in NUREG/CR-4686
- Valve and Snubber Operability and Piping Support Fragility Experiments Performed at the Heissdampfreaktor, to be reported in two forthcoming NUREG reports
- Nuclear Containment Piping Penetrations and Isolation Valve Response to Design Basis and Severe Accident Loadings, reported in NUREG/CR-5043
- Gate Valve Closing Requirements in a High Energy Boiling Water Reactor (BWR) Line Break (GI-87),<sup>b</sup> to be reported in a forthcoming NUREG report.

## NUCLEAR CONTAINMENT PURGE AND VENT VALVES CLOSING IN A DESIGN BASIS LOCA ENVIRONMENT

Nuclear containment purge and vent systems are open loop systems where the containment atmosphere communicates directly with the atmosphere outside containment. Such systems are typically isolated by two rubber sealed

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a. Work supported by the U.S. Nuclear Regulatory Commission, Division of Engineering, Office of Nuclear Regulatory Research, under DOE Contract No. DE-AC07-76ID01570; Dr. G. H. Weidenhamer, Technical Monitor.

b. "Failure of HPCI Steam Line Without Isolation" (Generic Issue 87).

butterfly valves. The concern over the the ability of these valves to close under design basis LOCA conditions was identified in the Three Mile Island Action Plan, NUREG-0660, Item 11.E.4.2.

For this test program, conducted in 1984, three valves representative of nuclear containment purge and vent isolation valves were selected for testing at typical predicted plant accident pressures. Two were 8-in. valves typical of two different internal butterfly designs, and the other was a 24-in. valve of the same design as one of the 8-in. valves. This provided the opportunity to compare design differences on valve response and to investigate valve response extrapolation equations. Before this test program, only very small valves had been tested with compressive fluids under pressure, and those were tested at relatively low pressures.

The valves were individually installed in test loops and subjected to full flow closing tests. The upstream pressure on the valves was maintained at a constant level throughout the valve closure. Parametric studies included varying the test pressures from 5 to 60 psig and varying the valve orientation and upstream piping geometry. The valve discharge was through an instrumented test section to atmosphere. Figure 1 shows the valve installed in the normal test loop, and Figure 2 shows the valve moved upstream to an elbow. These asymmetric closing, high-aspect-ratio valves required six test positions (shown in Figures 1 and 2) to completely characterize valve response in all possible plant installation orientations.



Figure 1. Valve positions with uniform inlet flow and counterclockwise (CCW) disc rotation on closure.

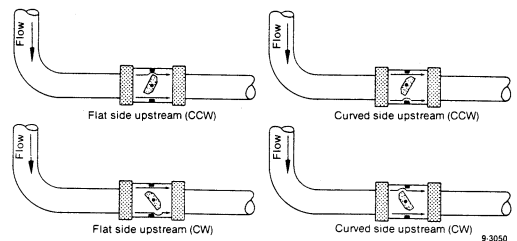


Figure 2. Valve positions with nonuniform inlet flow and either clockwise (CW) or CCW disc rotation on closure.

Test results provided evidence that several of the terms used in the industry's equations for valve closing torque requirements and extrapolation of small valve torques to predict large valve torques were not conservative. Valve installation orientation and upstream piping geometry significantly affected the amount of torque needed to close the valve. Typically, the closing torque was higher with the curved side of the disc upstream than for the same valve orientation with the flat side upstream. Piping elbows immediately upstream of the valve increased torque demands up to 1.5 times that required in a straight pipe. It was also determined that the elastomer sealing materials were very near their material property limits at design basis accident temperatures and pressures; in a plant installation, some leakage could be expected from the first isolation valve.

The results of this work were used by the NRC as part of the technical basis for reviewing utility submittals on TMI Action Plan Item 11.E.4.2 and for resolving the action item. The work additionally formed the basis for a change in the flow interruption test annex of QV-4, the ASME draft standard for valve qualification.

**VALVE AND SNUBBER OPERABILITY AND PIPING SUPPORT FRAGILITY EXPERIMENTS PERFORMED AT THE HEISSDAMPREAKTOR**

The INEL participated in two internationally sponsored seismic research programs conducted at a decommissioned reactor facility, the Heissdampfreaktor (HDR), located in the Federal Republic of Germany. INEL researchers modified an existing full scale HDR piping system called the Versuchskreislauf (VKL) (Figure 3) by installing (a) a refurbished, naturally aged 8-in. motor-operated gate valve that had served 25 years in the secondary system at the Shippingport Atomic Power Station and (b) a piping support system, including snubbers and rigid struts, typical of U.S. nuclear design.

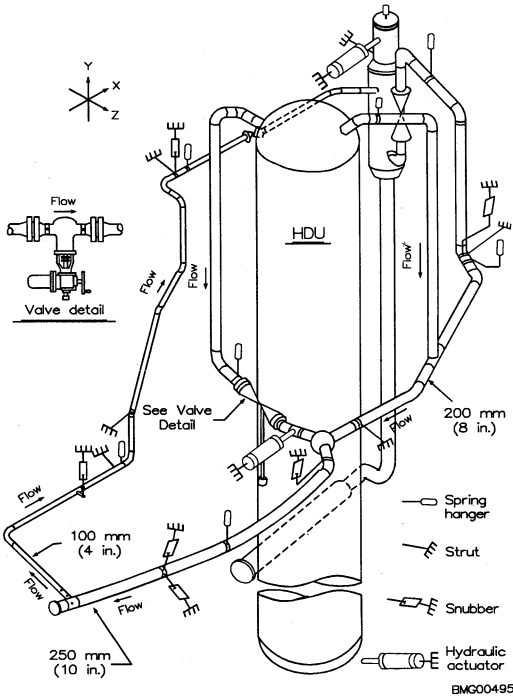


Figure 3. The VKL with servo-hydraulic shakers and other equipment installed for the SHAM test series.

0.6 g to more than 5 g zero period acceleration. During the tests with the highest inputs, local strains in excess of 0.9% were measured in highly stressed components. In all, 51 tests were performed on six piping support systems during the SHAM test series.

The piping system was water-filled and pressurized for both research programs. Moderate flows were established for some tests, and for one series of tests the internal fluid was heated to BWR primary temperature/pressure conditions.

Analyses of the data from the SHAG program did not identify any significant problems associated with the seismic loading. The results did show that modeling and predicting the seismic loads on an in situ piping system are very difficult. Typically, loads at piping supports were underpredicted, and piping strains were overpredicted. The amplified response of the 8-in. gate valve, particularly the response measured at the motor operator, was considerably different than that determined from seismic bench testing.

The two seismic research programs, called SHAG and SHAM, differed in the method and magnitude of piping and valve excitation. In the SHAG program, conducted in 1986, the entire containment building was excited by a large, twin arm, rotary mass, coastdown shaker mounted on the operating floor (30 m level) of the reactor building. The shaker transmitted force and motion from the building floors and internal structures to the piping systems and components in the building. Twenty-five tests were performed with initial starting frequencies from 1.6 to 8 Hz. Seven different piping support systems were used, providing piping system and valve responses from very flexible, moderately flexible, and very stiff support systems. Inputs to the building exceeded 0.3 g, and piping system responses exceeded 3 g.

In the SHAM program, conducted in 1988, two 40-metric-ton servohydraulic shakers were attached directly to the VKL piping system. Earthquake-like time histories were input to the shakers. Shaker outputs to the piping system varied in magnitude from

The valve experienced operability problems not related to the seismic loads. During some of the tests, the motor operator failed to trip the torque switch on closing. Data analyses and follow-on testing identified three problems that have generic implications for dc-powered motor operators.

First, resistance in the electrical circuit supplying power to the motor operator can significantly reduce the output torque of high torque, intermittent duty dc motors. The current stamped on the name plate is the continuous duty current, which is 20% of that required to produce the maximum rated torque. Cabling sized according to the name plate current will not supply adequate current for low speed, high load applications. Depending on the circuit design, four long cable runs, not two, can contribute to the circuit resistance. Resistance of only one ohm can reduce the output torque by one third to one half at motor stall. The back electromotive force generated by a dc motor at the low loads normally associated with in-plant testing can prevent detection of high resistance in the power cabling (IE Bulletin 80-11).

Second, dc motors under high loads heat up very rapidly because of high current at low speed. Motor heating increases resistance in the motor windings and thus also reduces motor torque. High ambient temperatures that might accompany an accident can further reduce motor torque output. On November 3, 1988, Limitorque filed a 10 CFR part 21 report addressing heating of dc motors.

Third, aging of the helical torque spring used in the older Limitorque SMA motor operators can cause the motor operator to trip the torque switch prematurely. Investigations showed that in the HDR test valve and in at least one other similar valve, the torque spring had taken a permanent set over time, so that the output torque for a given torque switch setting was less than specified. (An NRC information notice is currently being drafted to address this concern.)

Analyses of SHAM testing results are still in progress; early findings show that in situ piping systems subjected to simulated seismic events are very rugged. The valve motor operator experienced responses in excess of 11 g without operational problems. The safety margins of rigid struts exceeded five times the manufacturers' ratings. Two snubbers of the same size and make failed at loads lower than their rated load. Typically, the other snubbers failed at loads over three times their rated loads. The experimental program also identified loss of preload tension as the first precursor to concrete anchor failure. Failure analysis of the snubbers that failed at loads lower than design is scheduled for the second quarter of FY 89.

## **CONTAINMENT PIPING PENETRATIONS RESPONSE IN AN ACCIDENT ENVIRONMENT**

The Containment Piping Penetration research program, conducted in 1987, subjected three typical light water reactor containment piping penetration systems and their isolation valves to the mechanical, thermal, and pressure loads of design basis and severe accident loadings. The three systems tested were (a) an 8-in. gate valve system modeling a containment spray system, (b) an 8-in. butterfly valve system modeling a purge and vent valve system, and (c) a 2-in. globe valve system modeling the numerous small-bore piping systems that typically penetrate the containment. The valve types, valve sizes, piping configurations, penetrations, and supports used for this test program were typical of those used for containment isolation in commercial U.S. nuclear power plants.

The three systems were individually mounted in a test fixture (Figure 4). In initial tests, the systems were subjected to the loads expected from a containment response to a design basis loss-of-coolant accident. In later tests, the loads were increased to simulate severe accident responses. The mechanical loads on the piping system were applied in a step-like fashion; a