UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

November 30, 1993

NRC INFORMATION NOTICE 93-88: STATUS OF MOTOR-OPERATED VALVE PERFORMANCE PREDICTION PROGRAM BY THE ELECTRIC POWER RESEARCH INSTITUTE

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to alert addressees to preliminary results of motor-operated valve (MOV) tests conducted by the Electric Power Research Institute (EPRI). It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Background

On June 28, 1989, the NRC issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," to request that nuclear power plant licensees and construction permit holders verify the design-basis capability of their safety-related MOVs. In GL 89-10, the NRC staff requested that licensees and permit holders test each MOV within the scope of the generic letter under design-basis differential pressure and flow conditions, where practicable. The recommended schedule in GL 89-10 would have licensees and permit holders verify MOV design-basis capability by June 28, 1994, or three refueling outages after December 28, 1989 (whichever is later).

In response to concerns regarding MOV performance, EPRI and its utility advisors established a research program to develop a methodology to predict the performance of MOVs under design-basis conditions. NUMARC coordinates the interaction between EPRI, its utility Technical Advisory Group (TAG), and NRC staff related to the EPRI program. The EPRI program includes detailed analyses and testing of MOVs at test facilities and nuclear power plants. The EPRI MOV Performance Prediction Methodology is intended to allow licensees to demonstrate the design-basis capability of MOVs based on analytical predictions combined with diagnostic tests conducted under static conditions. In August and October 1993, EPRI presented the status and preliminary results from its Flow Loop Testing Program to the NRC staff. The flow loop results in the EPRI presentation have not received full quality assurance verification, but the preliminary information may be helpful to licensees as they implement their MOV programs.

In a letter on September 16, 1993, NUMARC provided responses from EPRI to NRC staff questions on the EPRI MOV Performance Prediction Program. Among the information provided in the enclosure to the letter, EPRI stated that its program is expected to cover about 90 percent of gate valves (about half with its computer code and half with empirically-based data), essentially all globe valves, and about 95 percent of butterfly valves. The globe and gate valves are covered primarily by the computer code. EPRI also stated its method for determination of operator output torque capability under degraded voltage conditions is to apply standard methods as documented in EPRI NP-6660-D (March 1990), "Application Guide for Motor-Operated Valves in Nuclear Power Plants."

Description of Circumstances

In conducting its MOV Performance Prediction Program, EPRI tested 28 gate, 4 globe, and 2 butterfly valves under a total of 62 test conditions. These tests were performed at Wyle Laboratories and Siemens test facilities. EPRI plans to obtain test data for an additional 35 valves being tested in nuclear power plants. In addition, EPRI completed testing at Kalsi Engineering of 10 butterfly valve designs to assess flow and upstream piping configuration effects. The results summarized below are based on the Wyle/Siemens MOV tests.

1. Gate Valves

EPRI stated that all gate valves tested were initially preconditioned by conducting a large number (50-1000) of short (no flow) strokes in cold water under differential pressure loading. Initial "sliding friction coefficients," prior to preconditioning, generally ranged from 0.2 to 0.4. EPRI indicated that, after preconditioning, "apparent friction coefficients" ranged from 0.3 to 0.6 for all but four valves tested under cold water pumped-flow conditions. The "apparent friction coefficients" for the remaining four valves ranged from 0.66 to 1.93. EPRI results demonstrated "apparent friction coefficients" ranging from 0.34 to 0.41 for hot water pumped-flow conditions, 0.35 to 0.8 for hot water blowdown conditions, and 0.25 to 0.64 for steam blowdown conditions. EPRI's "apparent friction coefficients" reflect all valve internal phenomena and are not necessarily indicative of a "sliding friction coefficient." The major difference between the "apparent friction coefficient" used by EPRI and the "valve factor" used historically by valve vendors in sizing motor operators is the consideration of the valve disc angle in determining the EPRI "apparent friction coefficient."

Most valve vendors have used a "valve factor" of 0.3 for flexible wedge gate valves and 0.2 for parallel disc gate valves in sizing motor operators. Therefore, the EPRI test results indicate that the thrust required to operate gate valves could be significantly greater than the thrust predicted by the valve vendors. The EPRI blowdown test results are generally consistent with those obtained in the limited testing program conducted by the Idaho National Engineering Laboratory (INEL) for the NRC Office of Nuclear Regulatory Research in 1989.

EPRI reported that the valve sliding friction coefficient tends to decrease with increasing differential pressure which lends support for linear extrapolation of reduced differential pressure results when there is a low potential for valve damage (for example, under nominal flow velocity pumped-flow conditions).

EPRI reported that several gate valves were damaged during hot water and steam blowdown testing. These included a 6-inch Anchor-Darling valve (disk and seat damage); a 6-inch Crane valve (guide damage); a 10-inch Velan valve (guide damage); a 6-inch Walworth valve (guide damage); and a 10-inch Edward valve (disk and seat damage). Two of the damaged valves exhibited "apparent friction coefficients" exceeding 0.6.

Two gate valves were damaged under cold water pumped-flow conditions. These included a Velan 6-inch valve (plastic bending of body guides at high flow velocity greater than 30 feet per second) and an 18-inch Anchor-Darling valve (valve disk forced through seating area resulting in leakage above disk).

EPRI test results revealed that it is generally not possible to determine accurately the point of flow isolation prior to disk wedging based on the thrust diagnostic trace alone.

EPRI stated that it had not observed differences in thrust requirements for valve operation between valves installed in horizontal pipes with the stem either vertical or horizontal. This finding differs from some operating experiences in nuclear power plants.

2. Globe Valves

EPRI stated that, for incompressible flow conditions, globe valve thrusts are consistent with industry calculational-method predictions only if the appropriate area is chosen for differential pressure application. The appropriate area (disk mean seat area versus disk guide area) appears to be unique to valve design. It was determined that use of disk mean seat area rather than disk guide area can result in significant underestimation of required thrust for some globe valve designs. Specifically, one globe valve tested under cold water pumped-flow conditions required approximately twice as much thrust to close using disk mean seat area and a valve factor of 1.0.

A two-inch Rockwell/Edward globe valve, tested under hot water blowdown conditions, exhibited thrust requirements exceeding predictions based on disk guide area by approximately 35 percent. This valve sustained damage to the portion of the body bore that guides the disk.

Current industry practice for determining the required thrust for globe valves varies by manufacturer. Many manufacturers assume disk mean seat area multiplied by a valve factor in the 1.0 to 1.1 range. Others use disk guide area in making thrust predictions. Therefore, the EPRI results indicate that actual thrust requirements may exceed those predicted using current industry practice for some globe valve designs.

3. Butterfly Valves

EPRI stated that the Wyle flow loop testing revealed torque requirements to operate Pratt butterfly valves which were bounded by the most current torque predictions of the manufacturer. However, butterfly valves at some nuclear power plants (for example, Catawba and Palo Verde) have demonstrated torque requirements that exceed vendor predictions. EPRI is currently evaluating data from testing conducted at Kalsi Engineering to assess butterfly valve torque requirements for a wide range of butterfly valve designs.

4. Data Interpretation and Assessment

In July 1993, EPRI sent a Quarterly Status Report to all utilities participating in the EPRI MOV Performance Prediction Program. This report summarized preliminary flow loop test results. After the completion of Wyle/Siemens quality assurance checks, EPRI plans to update this information in its next Quarterly Status report scheduled for late 1993. Detailed test reports documenting these results are scheduled for delivery to participating utilities between October and December 1993. EPRI stated that, in interpreting the EPRI flow loop test results, utilities need to understand the assumptions and equations that were used by EPRI in presenting the data. For example, the EPRI calculated "apparent friction coefficient" for gate valves is based on the equation provided in EPRI Report NP-6660-D, "Application Guide for Motor-Operated Valves in Nuclear Power Plants." This equation is solved for "apparent friction coefficient" using (1) the maximum measured stem thrust which occurs prior to the initiation of wedging (for valve closing) or the maximum thrust which occurs after cracking (for valve opening); (2) full (valve closed) tested differential pressure; (3) mean seat area; (4) valve disk angle; (5) full (valve closed) upstream tested pressure for stem rejection thrust; and (6) measured values of packing load.

EPRI stated that valve design and test conditions, maintenance history, and operating experience may be important in assessing the applicability of EPRI test results to plant MOVs.

EPRI uses the greatest thrust requirement to overcome differential pressure and flow to determine its "apparent friction coefficient." EPRI assumes the highest differential pressure observed during the test regardless of the stem position where the greatest differential pressure/flow required thrust occurs. This results in a lower calculated friction coefficient than would be determined if the actual differential pressure at the point of greatest thrust was used in determining the friction coefficient.

EPRI plans to submit sections of a topical report for NRC review as they are completed between November 1993 and April 1994. EPRI intends to submit supporting reports in advance of the final topical report to allow the staff to raise questions with EPRI early in the review process.

EPRI stated that its final methodology is scheduled for delivery to utilities in April 1994 as a tool that may be used to confirm many aspects of MOV calculations and setup. Further, the EPRI flow loop test results provide licensees with information which might be helpful in supplementing other "best available" data in establishing MOV switch settings.

Discussion

Since EPRI initiated its MOV Performance Prediction Program, the NRC staff has conducted public meetings with NUMARC and EPRI to discuss the goals of the EPRI program, the development of the program activities to accomplish those goals, the tests conducted in support of the program and the results of those tests, and the completion schedule for the program. The staff has provided questions and comments to NUMARC and EPRI on the EPRI MOV program as a result of these meetings. For example, in a public meeting on October 6-7, 1993, the staff emphasized the need for EPRI to ensure that licensees clearly understand the application of the EPRI test data and methodology. Also at this meeting, contents of this notice were discussed and the comments from EPRI have been considered. The staff expressed concern about the valves damaged during the EPRI testing and the apparent lack of action by some valve manufacturers in response to the valve damage. The staff also discussed the need for EPRI to ensure that adequate peer review of the EPRI methodology is conducted.

Although some issues remain to be resolved, the EPRI testing program should provide a significant amount of MOV test data that can assist nuclear power plant licensees in demonstrating the design-basis capability of MOVs that cannot be tested under dynamic conditions as installed. The preliminary test information provided in this notice is provided for licensee consideration in implementing programs in response to GL 89-10 . The staff plans to conduct additional public meetings with NUMARC and EPRI to discuss the status of the EPRI MOV program. The staff will consider the need for additional generic communications to nuclear power plant licensees and construction permit holders as additional information is obtained from the EPRI MOV program.

Related Generic Communications

- NRC Information Notice 90-40, "Results of NRC-Sponsored Testing of Motor-Operated Valves."

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the person listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contact: Thomas G. Scarbrough, NRR
(301) 504-2794

Attachment:
List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-87	Fuse Problems with Westinghouse 7300 Printed Circuit Cards	11/04/93	All holders of OLs or CPs for nuclear power reactors.
93-86	Identification of Isotopes in the Production and Shipment of Byproduct Material at Non-power Reactors	10/29/93	All holders of OLs or CPs for test and research reactors.
93-85	Problems with X-Relays in DB- and DHB-Type Circuit Breakers Manufactured by Westinghouse	10/20/93	All holders of OLs or CPs for nuclear power reactors.
93-84	Determination of Westinghouse Reactor Coolant Pump Seal Failure	10/20/93	All holders of OLs or CPs for pressurized water reactors (PWRs).
93-83	Potential Loss of Spent Fuel Pool Cooling Following A Loss of Coolant Accident (LOCA)	10/07/93	All holders of OLs or CPs for boiling water reactors (BWRs).
93-82	Recent Fuel and Core Performance Problems in Operating Reactors	10/12/93	All holders of OLs or CPs for nuclear power reactors and all NRC-approved fuel suppliers.
93-81	Implementation of Engineering Expertise on Shift	10/12/93	All holders of OLs or CPs for nuclear power reactors.
93-80	Implementation of the Revised 10 CFR Part 20	10/08/93	All byproduct, source, and and special nuclear material licensees.
93-79	Core Shroud Cracking at Beltline Region Welds in Boiling-Water Reactors	09/30/93	All holders of operating licenses or construction permits for boiling-water reactors (BWRs).

OL = Operating License
CP = Construction Permit

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orig /s/'d byBKGrimes

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OFFICE	RPB:ADM	OGCB:DORS:NRR	C/OGCB:DORS:NRR	D/DORS:NRR
NAME	RSanders*	JBirmingham*	GHMarcus*	BKGrimes
DATE	11/18/93	11/22/93	11/23/93	11/24/93

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