NUREG/CR-5406
EGG-2569
Vol. 3

BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test

Review of Issues Associated with BWR Containment Isolation Valve Closure

Prepared by K.G. DeWall, R. Steele, Jr.

Idaho National Engineering Laboratory

EG&G Idaho, Inc.

Prepared for
U.S. Nuclear Regulatory Commission

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW, Lower Level, Washington, DC 20555

2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082

3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG/CR-5406
EGG-2569
Vol. 3
R1

BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test

Review of Issues Associated with BWR Containment Isolation Valve Closure

Manuscript Completed: September 1989
Date Published: October 1989

Prepared by
K.G. DeWall, R. Steele, Jr.

Idaho National Engineering Laboratory
Managed by the U.S. Department of Energy

EG&G Idaho, Inc.
P. O. Box 1625
Idaho Falls, ID 83415

Prepared for
Division of Engineering
Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN A6857, B5529

ABSTRACT

This report presents the results of research performed to develop technical insights for the NRC effort regarding Generic Issue 87, "Failure of HPCI Steam Line Without Isolation." Volume III of this report contains the data and findings from the original research performed to assess the qualification of the valves and reported in EGG–SSRE–7387, "Qualification of Valve Assemblies in High Energy BWR Systems Penetrating Containment." We present the original work here to complete the documentation trail. The recommendations contained in Volume III of this report resulted in the test program described in Volume I and II. The research began with a survey to characterize the population of normally open containment isolation valves in those process lines that connect to the primary system and penetrate containment. The qualification methodology used by the various manufacturers identified in the survey is reviewed and deficiencies in that methodology are identified. Recommendations for expanding the qualification of valve assemblies for high energy pipe break conditions are presented.

A6322—Environmental Qualification of Mechanical and Dynamic Qualification of Mechanical and Electrical Equipment Program

EXECUTIVE SUMMARY

Volume III of this report discusses research performed to develop technical insights for the NRC effort regarding Generic Issue 87, "Failure of HPCI Steam Line Without Isolation." The work was performed under FIN A6322. The Office of Nuclear Regulatory Research sponsors the Mechanical Equipment Qualification Research Program (FIN A6322) and is assisting in the resolution of this issue.

Four BWR systems, the Emergency Cooling System, the High Pressure Coolant Injection System, the Reactor Core Isolation Cooling System, and the Reactor Water Cleanup System, were included in the valve assembly characterization. The "typical" containment isolation valve is a 3 to 10 in., 600 to 900 lb, gate valve. The most common design is a cast steel, flexible wedge, pressure–seal valve with a Limitorque operator (AC inside and DC outside of containment). The Anchor/Darling Valve Company manufactures approximately 40% of the valves identified.

The mitigation of a high energy pipe break is within the design basis for the above valve assemblies, with typical system design conditions of 1250 psi and 575°F. No flow testing has been performed under these conditions to verify the presumptions used by manufacturers in the qualification analysis calculations. Operator torque switch settings are determined using calculations supplied by the valve vendor, which could lead to inadequate torque settings to close the valve if the original calculations are not conservative.

Most of the valve and operator manufacturers use the same equation to size operators with minor variations in coefficients. In this equation, the required thrust to close the valve is equal to the sum of the disc drag load due to differential pressure, the stem end pressure load, and the packing drag load. The service conditions used in the thrust equation are supplied by each individual plant. Four areas have been identified as having the most influence on stem thrust requirements. Observations concerning these four areas are noted below.

1. Repeated cycling can have a significant effect on valve thrust requirements.

2. The typical industry 0.3 disc friction coefficient is not conservative for all cases.

3. Mass flow/momentum influence on valve thrust requirements may be significant.

4. Increased temperature causes a significant increase in valve closure loads

The limited number of tests performed to assess gate valve flow interruption capability with high pressure steam have resulted in a relatively frequent inability to isolate. The data now available suggest that industry may be using nonconservative friction factors and possibly under–estimating valve stem thrust requirements. Additional work is needed to determine whether present qualification practices are adequate.

CONTENTS

FIGURES

TABLES

ACKNOWLEDGMENTS

The author wishes to express his sincere appreciation to the following persons for their assistance in this research effort. Robert Bergman (Anchor/Darling Company), Paul Lawson (Crane Company), Robert Kestler (Wm. Powell Company), Birk Neilson (Velan Incorporated), John Ferrell (Velan Incorporated), and Cliff Johnson (Walworth Company) all contributed by providing information about their respective valve qualification methodology.

REVIEW OF ISSUES ASSOCIATED WITH BWR CONTAINMENT ISOLATION VALVE CLOSURE

1. INTRODUCTION

The United States Nuclear Regulatory Commission (USNRC) has assigned a "HIGH" priority to Generic Issue No. 87, "Failure of HPCI Steam Line Without Isolation."[1] The issue concerns a postulated break in the High Pressure Coolant Injection (HPCI) steam supply line in Boiling Water Reactors (BWRs) and the uncertainty regarding the capability of the HPCI steam supply line isolation valves to close under those conditions. A similar situation can occur in the Reactor Core Isolation Cooling (RCIC) System and the Reactor Water Cleanup (RWCU) System, along with other high energy steam lines coming off of the Main Steam Line (MSL). Without isolation, such breaks have high potential consequences because other emergency equipment located in the vicinity of the break would be exposed to an environment which could result in common-cause failure. To resolve this issue one must answer two questions: (a) have the subject valves been qualified for the conditions expected to result from a high energy pipe break and (b) were the methods used to qualify the valve assemblies adequate to assure operability under pipe break conditions.

The Office of Nuclear Reactor Regulation (NRR), Division of Safety Review and Oversight is coordinating the actions necessary to resolve this licensing issue and has requested assistance from the Office of Nuclear Regulatory Research (RES), within ongoing work on the Mechanical Equipment Qualification Research Program (FIN A6322). The Idaho National Engineering Laboratory (INEL) is the contractor for this program.

1.1 Background

The HPCI steam supply line typically has two containment isolation valves in series, usually one inside containment and one on the outside of containment. These valves are normally open in most plants. The HPCI supply valve, located adjacent to the turbine, and the turbine stop valve are normally closed. The RCIC and RWCU each have two isolation valves which are normally open. The RWCU valves must remain open if the system is to operate.

The gate valve is designed for use in a system where a positive shut-off is required with minimal pressure drop. It is ideally suited to those situations where isolation of one part of a system from another is required and control of the dynamic properties of the fluid (throttling) is unnecessary. With the disc (or gate) in the raised position, the run of the valve is free of any obstruction with approximately the same head loss as in the adjacent piping. When the disc is lowered into the seat, the upstream pressure forces it against the seat creating a seal and isolating the downstream system from the fluid. The thrust required to close and open the valve is not dependent on flow direction; however, the thrust requirement may be affected by the mass flow through the valve.

Failure to close, defined as the inability of the valve operator to move the gate from the full open to full closed position in the specified time duration, can result from many causes.[2] Under GI-87 concerns, the two most important reasons for valves failing to close are

1. Excess stem seal loads

2. Large pressure or flow induced forces.

The first of these, excess stem seal loads, most often result from pressing the stem packing too tightly against the stem by overtightening the packing compression bolts. This condition may develop during packing maintenance either inadvertently or in an attempt to overcome leaks due to stem scoring. Functional testing after maintenance is typically performed to guard against over tightening of the packing.

Large pressure or flow induced forces can occur when a valve must close to shut off flow from a downstream pipe break, precisely the concern of GI-87. Under these conditions, the flow through the valve can reach critical velocity as the valve closes. The result is large differential pressure and inertial/momentum load on the disc forcing the disc against the seat and increasing friction.

Due to flow limitations at the valve manufacturers' facilities, only the opening characteristics of the valve are typically tested under operating conditions. As part of the utilities' In–Service Testing Program, the operation of the valves is tested periodically but without steam flow. The capability of the valves to close when exposed to the forces resulting from a break downstream has typically not been fully tested.

2. OBJECTIVES

The overall purpose of the INEL research is to provide a technical basis for the resolution of GI-87. The following research objectives were developed to guide the research toward this end.

1. Identify (by manufacturer and model) the specific valve assemblies used in the BWR systems which fall under the concerns of GI-87. (Documented in "Summary of Valve Assemblies in High Energy BWR Systems Outside of Containment—Interim Report.")

2. Determine the conditions for which those valve assemblies have been qualified and identify valve assemblies that have adequate qualification to assure isolation of a high energy line break.

3. Review the qualification methods used by vendors and identify deficiencies in that methodology.

4. Recommend appropriate follow-up efforts required to assure adequate qualification of questionable valve assemblies.

3. GI–87 VALVE ASSEMBLIES

A review of available information sources to identify the systems applicable to GI–87 was performed and a determination was made as to the valve and operator manufacturers, types, and sizes used in those systems. The following is a summary of the results presented in Appendix A. The BWR systems containing isolation valves of concern under GI–87 are the following.

1. Emergency Cooling System (steam leaving the reactor—BWR–2 only)

2. High Pressure Injection System (HPCI) (turbine steam supply—BWR–3 and 4 only)

3. Reactor Core Isolation Cooling (RCIC) (turbine steam supply)

4. Reactor Water Cleanup (RWCU).

Preliminary and Final Safety Analysis Reports (PSAR/FSAR) and data from the Institute of Nuclear Power Operations (INPO) Nuclear Plant Reliability Data System (NPRDS) were used to determine specific valve assembly information. With only a few exceptions, the valves that must be qualified in order to resolve GI–87 are pressure–seal, cast steel, flexible wedge gate valves in the 3 to 10 in. range and 600 and 900 lb. class.

The most predominant valve manufacturer is the Anchor/Darling Valve Co. with 41% of the containment isolation valves. The other manufacturers are Borg–Warner (2%), Crane Co. (18%), William Powell Co. (11%), Velan Inc. (16%), and Walworth Co. (12%). Limitorque Corporation manufactures 94% of the valve operators. The remaining 6% are identified as Philadelphia Gear Corporation operators (predecessor of Limitorque Corporation).

4. PRESENT VALVE ASSEMBLY QUALIFICATION

The following paragraphs discuss the results of research performed to determine the conditions for which the valve assemblies identified in the previous section have been qualified and to determine the methods used by utilities and vendors to provide this qualification. An essential part of this discussion will be a review of operator sizing and torque switch setting practices, since these items directly control valve disc movement.

4.1 Valve Operating Design Basis

The second objective listed in Section 2 of this report is to determine the conditions for which the valve assemblies have been qualified and identify the valve assemblies that have adequate qualification to assure isolation of a high energy line break. In order to complete this objective, a number of utility submittals in response to IE Bulletin 85–03 (Reference 3) were reviewed to identify maximum valve design differential pressure and temperature. Although the bulletin addressed valve torque switch settings exclusively, the valve design information requested covers the containment isolation valves of interest to GI–87 in the steam lines for the HPCI and RCIC systems.

The design basis for each valve consists of (a) the maximum differential pressure expected during opening and/or closing of the valve for both normal and abnormal events, and (b) the temperature corresponding to these conditions. At most plants, the maximum expected differential pressure is conservatively considered to be the maximum upstream pressure. No credit is taken for the downstream pressure. Thus, the maximum expected differential pressure will be the most conservative enveloping differential pressure that could be experienced by the MOVs during various plant operational modes.

Of the plants responding to IE Bulletin 85–03, most identified the pipe break condition as a design basis event for the containment isolation valves in the HPCI and RCIC steamlines and the RWCU suction line. At the Perry Nuclear Power Plant for example, the control switch settings for these valves take into account line breaks, and are designed to provide positive valve actuation up to the maximum differential pressures expected to be seen across the valve in either the open or close direction during a design basis accident condition. This envelopes single equipment failure or inadvertent equipment operation.

The utility submittals in response to IE Bulletin 85–03 indicated that the upstream (and thus maximum differential) pressure ranged from 1100 to 1375 psig and the corresponding temperatures ranged from 540 to 585°F. The submittals verify the FSAR information found in the first part of the GI–87 study and lead to the conclusion that high energy pipe break isolation is within the valve's design basis.[4,5,6]

4.2 Utility Qualification Programs

Specific information on the valves identified in the GI–87 valve survey were obtained from a representative nuclear power plant. The system design pressures and temperatures, valve sizes, and valve and operator manufacturers at the plant are typical of the majority of operational BWRs.

The valves used in the HPCI, RCIC, and RWCU systems are manufactured by Anchor/Darling Co. and utilize Limitorque operators. They are of the same sizes, type, and class as those listed in Section 3. These flexible wedge gate valves consist essentially of a one piece wedge with the areas behind the seating surfaces hollowed out to allow more flexibility to conform to the seat alignment. The bodies of these valves have cast–in disc guides.

The purchase specifications and requirements include environmental conditions, thermal transients, and pressure, temperature, flow and differential pressure requirements.

The program for selecting correct valve switch settings consists of the following elements:

1. Calculation of design differential pressures during the preparation of equipment specifications.

2. Development of initial torque switch settings by the valve or motor-operated vendors.

3. Vendor testing of representative valves at design flows and differential pressures to verify adequate performance at the conditions specified in (1), and the switch settings selected in (2).

4. Stroke testing (with no differential pressure present) of all valves, using the Motor Operated Valve Analysis and Test System (MOVATS) to verify proper torque and limit switch settings.

The torque switch, limit switch, and stem packing adjustments are specified by the manufacturer in the Anchor/Darling Instructions for the Installation, Operation, and Maintenance Manual.

The following list details the testing that was performed.

1. Hydrostatic Testing

   a. Test is performed in accordance with the Code.

   b. The valve must be stroked six times following the hydrostatic test.

2. Functional Testing

   a. Valve is oriented for the most adverse conditions.

   b. The SSE deflection is imposed on the operator.

   c. Valve is internally pressurized to the maximum design pressure.

   d. The valve assembly is actuated using the minimum actuation supply voltage

   e. The valve must open and close within the specified time.

3. Seat Leakage

   a. Leakage shall not exceed two cc/hr per inch of nominal valve size.

   b. The duration of the test shall be at least four minutes.

The documentation from the representative BWR plant included a copy of a data sheet from a valve closure test conducted by Wyle Laboratories and a comparison of the design versus "realistic" valve movement torque requirements. The data sheet contained information gained as part of a Flow Interruption Capability Test and is reproduced in this report as Table 1. The only conclusions one can make are the following: (a) the test began with the system at the design pressure of 1370 psig and a differential pressure across the valve of zero, (b) the valve closed in 2.09 seconds with a final upstream pressure of 1205 psig and downstream pressure of 390 psig, and (c) the largest differential across the valve disc during closure was 815 psi. This is much less than the full system pressure one would expect given a GI–87 type pipe break immediately downstream of the valve. The information given did not indicate the presence of high fluid flow during the test. In all, the test provided insufficient information to assure valve operability under high energy pipe break conditions.

No tests or analysis under blowdown conditions were performed for these valves by the valve manufacturer.

Table 1. Flow interruption capability test

Valve Closing Time (sec)	Upstream Steam Pressure (psig)	Downstream Steam Pressure (psig)	Test Valve Differential Pressure (psid)
0	1370	1370	0
0.1	1365	1365	0
0.2	1345	1345	0
0.3	1325	1325	0
0.4	1320	1320	0
0.5	1305	1305	0
0.6	1280	1280	0
0.7	1270	1270	0
0.8	1260	1260	0
0.9	1250	1250	0
1.0	1240	1240	0
1.1	1225	1225	0
1.2	1215	1215	0
1.3	1205	1195	10
1.4	1195	1175	20
1.5	1190	1140	50
1.6	1195	1095	100
1.7	1200	1025	175
1.8	1215	910	305
1.9	1230	770	460
2.0[a]	1220	560	660
2.1	1205	390	815
2.2	1220	270	950

a. NOTE: Valve closed at 2.09 seconds.

4.3 Vendor Qualification Methodology

The vendors of the most commonly used components were contacted and the utility submittals in response to IE Bulletin 85–03 were reviewed to better understand vendor qualification methodology, and to identify possible flow interruption test data sources. The results of this review are described below.

A gate valve operator must overcome a force equal to the differential pressure times a coefficient of friction (generally 0.3 for a wedge type gate and 0.2 for a parallel seat gate). Figure 1 shows a cutaway of a typical motor operated gate valve.[7,8] The equation used throughout most of the valve and operator sizing literature equates the closing stem thrust to the disc friction load plus the stem end load plus the packing drag load, as detailed in Equation (1).

$$T = \mu F_d + F_s + F_p \qquad (1)$$

where

T = required stem thrust

μ = Seat coefficient of friction

F_d = Disc differential pressure load

F_s = Stem pressure end load

F_p = Packing drag load.

The exact equation used by each vendor is proprietary as is the seat coefficient of friction. One vendor, however, uses the following equation instead of Equation (1).

$$T = [\mu F_{area} + F_{seat} + F_{stem}]\Delta P + F_p \qquad (2)$$

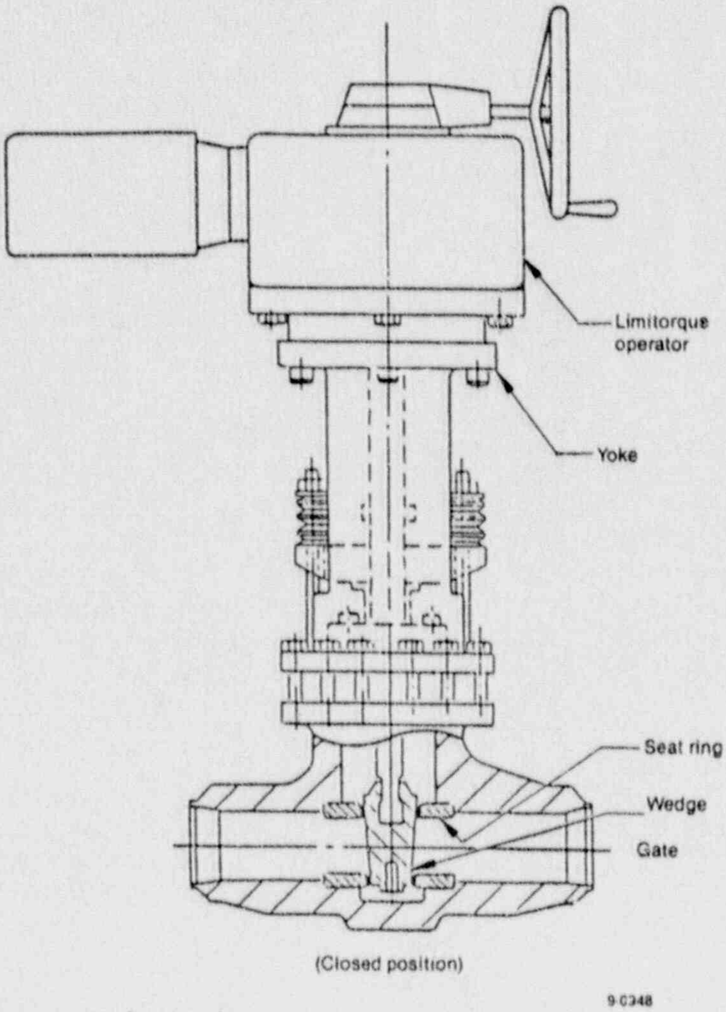

Figure 1. Typical motor-operated gate valve.

where

$T =$ Required stem thrust

$\mu =$ Seat coefficient of friction

$F_{area} =$ Area factor

$F_{seat} =$ Seat factor

$F_{stem} =$ Stem factor

$\Delta P =$ Differential pressure

$F_p =$ Packing drag load .

The seat coefficient of friction used for wedge–type gate valves in Equation (2) is 0.2. Equation (2) is based on seating and unseating loads.

Valve vendors place varying emphasis on the importance of other phenomena in their methods of determining valve thrust. The majority of the vendors do not take into account the effects of valve cycling, mass flow, and temperature. The only testing performed are the standard tests outlined in the ASME B&PV Code (pressure and shell tests). They believe that mass flow through the valve does not produce a significant disc load and consider only differential pressure effects on the gate. All of the valve operator sizing equations are proprietary including the disc friction factor. Most believe that valve opening loads exceed and will therefore bound closing loads.

A minority of the valve vendors take a different stand. Two vendors have observed instances where the valve thrust increased with cycling, attributed to temperature effects on the valve and operator, fluid type, valve design, and packing design. Their equations include additional force terms to account for mass flow through the valve and closing load versus opening load. The only vendor with high energy flow test experience has observed that, as the valve closes the mass flow through it adds a significant force resisting valve closure. For this reason closing thrust requirements are greater than opening thrust requirements for the same differential pressure across the gate. The gate friction factor used varies with conditions and valve design over a wide range in contrast with a single value used by most other vendors.

Under–estimating the valve loads stated above will most often result in an undersized ·· or operator or low torque switch settings on the motor operator. The General Electric Company standard design (for the newer BWR plants) for the motor–operated valves used in the systems of interest under GI–87 employs the following control switch scheme. In the opening direction a position limit switch contact is used to control valve stroke. The use of only a limit switch contact eliminates the possibility of the valve not opening on demand due to an incorrectly set torque bypass switch. In the closing direction both torque and limit switches, connected in a parallel arrangement are used to control valve stroke. This arrangement allows for positive valve closure by using the limit switch to control valve disc movement until the point just prior to disc seating. At this point the limit switch drops out of the circuit and the torque switch controls disc seating thereby preventing valve disc damage due to overtorquing the disc into the valve's seat. For the older BWR plants, the torque switch controls valve movement throughout the entire closure stroke.

As part of the valve procurement process, the vendors were required to prove valve operability at maximum system pressures. Flow interruption tests of valves in the size range of interest to GI–87 are very expensive, time consuming and require a large flow facility. As indicated above, only one vendor uses equations backed by actual flow interruption testing. In this case a 14 inch gate valve was tested with steam flow. All details about the test are considered proprietary by the vendor.

The remainder of the vendors use a substitute test to prove valve operability. In this test the valve is closed and full pressure is applied across the gate. The valve is then opened, the inference being made that the thrust required to open the valve is greater than that required to close it. The argument for this is that the pressure drop across the gate while closing off flow to a broken pipe cannot exceed the full pressure and therefore the valve's capability to close is demonstrated.

5. WEAKNESSES IN VENDOR METHODOLOGY

5.1 EPRI Marshall Test Program Results

Recent test programs suggest that the simplified approach described in the previous section may not be justified. In 1980 the Electric Power Research Institute (EPRI), on behalf of the participating PWR owners, conducted full flow steam testing on seven typical PWR PORV Block Valves at Duke Power's Marshall Steam Station. The results of this testing are described in the "EPRI/Marshall Electric Motor Operated Valve (Block Valve) Interim Test Data Report."[9]

The project objectives were to obtain preliminary information on electric motor operated valves by performing full flow steam testing. All seven PORV Block Valves tested were 3 inch 1500 lb class gate valves of similar design to those identified for GI-87. The valves were instrumented to measure motor current and valve stem position. Fluid pressures and temperatures were determined from instruments in the test piping; valve inlet temperature and body temperature were not monitored. Valve stem strain gauges were installed on the Westinghouse valves at special request from Westinghouse.

Three manufacturers' valves (Velan, Borg-Warner, and Rockwell International), as supplied, met the desired acceptance criteria during the test program. The valve assemblies fully closed and opened with little seat leakage for full flow and differential pressure conditions. One manufacturer's valve (Anchor/Darling) failed to close during preevaluation testing with the supplied operator. Excessive seat leakage was also observed. The valve was returned to the manufacturer where the seats were modified to increase the seat area, the valve stem and bonnet replaced, and a modified operator of the same model was installed. Retesting with the modified valve and operator still indicated closure trouble, so a larger operator capable of greater torque was installed. The valve assembly was successfully tested. This operator was then replaced with an operator of the same size as originally supplied. After verification of correct operator to stem alignment and setting of the closing torque switch settings to approximately maximum, the valve closed completely under full flow test conditions.

A second manufacturer's valve (Westinghouse) also experienced closure failures on two different models tested. Testing indicated that the Model 3GM88 valve with the vendor-recommended operator and torque switch setting was insufficient to reliably close the valve. Increasing the closing torque switch setting allowed the valve to completely close reliably with little or no seat leakage for the full flow steam test conditions. The model 3GM99 valve with the recommended operator and torque switch settings would not completely close the valve under full flow conditions. Based on valve stem strain measurements, a larger operator was installed and the valve passed the EPRI/Marshall testing sequence (the Model 3GM99's operator was also rewired to close using the close limit switch instead of the close torque switch). Additional testing was performed with the larger operator rewired in its normal mode, i.e., to deenergize the motor on the close torque switch. Again, the valve did not close completely under full flow conditions.

Table 2 presents a matrix of the valves tested versus the operators used and indicates whether they completely opened or closed. Valve functionability was successfully demonstrated for three of the five valve manufacturers, even though the valves with closure problems used equivalent operators. Stem load is then a function of not only the fluid conditions but also the valve design (i.e., wedge seat, materials, surface finishes, guilding, etc.). It is evident that , for some valve manufacturers, the actual stem load required to close the valve is quite different from the calculated stem load used for sizing the operators. All failures occurred during the closing cycle, casting serious doubt on the appropriateness of using valve opening tests at full differential pressure to prove closing cycle operability in a pipe break environment.

Table 2. Valve/operator matrix[10]

	Operator											
	Limitorque				Rotork							
	SB–00–15		SMB–000–10		14–NA1		16–NA1		16–NAX1		30–NA1	
Valve	Open	Close	Open	Close	Open	Close	Open	Close	Open	Close	Open	Close
Velan B10–3054B–13MS	Yes	Yes	—	—	—	—	—	—	—	—	—	—
Westinghouse 3GM88	Yes	Yes	—	—	Yes	No[a]	—	—	Yes	No[b]	—	—
Westinghouse 3GM99	Yes	No[c]	Yes	No[d]	—	—	—	—	—	—	—	—
Anchor/Darling Double–Disc 5J–1512	—	—	—	—	—	—	Yes	No[e]	—	—	Yes	Yes
Borg Warner 79294	—	—	—	—	—	—	—	—	Yes	Yes	—	—
Rockwell Inter. 1309460	—	—	Yes	Yes	—	—	—	—	—	—	—	—
Velan B10–3054B–13MS	—	—	Yes	Yes	—	—	—	—	—	—	—	—

a. The valve did not completely close on demand with the Rotork 14–NA1 operator. The Rotork 14–NA1 was substituted for the Limitorque SB–00–15 since the SB–00–15 was not electrically compatible with the Marshall Facility.

b. The Rotork 14–NA1 was replaced by a Rotork 16–NAX1. The valve completely closed on demand only when the torque switch was bypassed.

c. The Limitorque SB–00–15 replaced the SMB–000–10. The valve completely closed on demand only when the SB–00–15 was rewired to close using the limit switch instead of the torque switch.

d. The Limitorque SMB–000–10 did not completely close the valve on demand.

e. The Rotork 16–NAX1 even when modified could not completely close the valve on demand.

5.2 Westinghouse Test Program Results

At the completion of the EPRI/Marshall test program, Westinghouse conducted additional testing on the Westinghouse electric motor operated valves. The "EPRI Summary Report: Westinghouse Gate Valve Closure Testing Program," contains the results of this test program. [11] Although Westinghouse valves were not identified in the valve survey and are probably not used in BWR plants, they are similar enough in design to those valves utilized in BWR systems to make the following information of generic importance to GI–87.

To determine the causes of the higher than expected stem thrust measured during previous tests, the Westinghouse Electro–Mechanical Division undertook three testing programs.

1. A series of 50 separate water flow tests were conducted against 60 to 600 gpm flow and 1500 to 2600 psi differential pressure.

2. A mechanical fixture test was conducted using a hydraulic cylinder to apply simulated flow loads to the valve disc.

3. Friction factor tests were performed, utilizing small samples cut from the faces of actual discs and seat rings.

The test results indicate that:

1. The friction factor at room temperature will increase from as low as 0.12 until a level of 0.4 to 0.75 is reached at 100 to 200 cycles.

2. The magnitude of the friction factor at 180°F is higher than at room temperature with peak values of 0.64 to 1.00.

3. Dry data indicates little change in friction factor occurs with cycling, and that the friction level is approximately 0.3.

4. With 550°F steam, at a %0.1–inch stroke length, the friction factor starts in the 0.5 to 0.6 range and drops quickly to approximately 0.35.

5. Pause time under load (closed position) increases the friction factor, while pause time under no load (open position) decreases the friction factor.

As a result of the EPRI/Marshall, Almarez, and Westinghouse test programs, Westinghouse concluded that the valve closure problems were the result of under–predicting the friction load and therefore under–estimating the stem thrust required to close the valve against high differential pressures. Although tests showed friction factors ranging from 0.1 to 1.0, Westinghouse recommended that a friction factor of 0.55 be used in Equation (1).

It should be noted that several of the other valves in the EPRI program closed successfully even though their operators were most likely sized using the 0.3 disc friction factor. Westinghouse explained this as most likely resulting from the difference in operator sizing philosophy between Westinghouse and most other valve companies. Most other companies allow Limitorque Corporation to perform their operator sizing. Westinghouse suggests that the standard Limitorque technique may have sufficient margin built into it at other points of the sizing calculation that the final operator size is adequate and most valves would close at the higher actual loads. These added margins can result in operator stall output loads that can damage a valve not designed to accept them. Westinghouse attempted to minimize the potential for damage by reducing operator margins, making the Westinghouse design less tolerant of under–estimation of closing thrusts.

5.3 Ontario Hydro Flow Test

An additional valve flow interruption test program has been performed. The bulk of the test results are proprietary, however, a few general results are available.

Ontario Hydro performed a flow interruption test of an 8 inch, 900 lb, wedge type gate valve with an electrical motor operator. The test was performed for New Brunswick Power, at the Ontario Hydro Nuclear Process Components Testing Facility in Toronto, Ontario, Canada.

The valve test was a blow-down type test, with saturated water at approximately 525°F flashing to steam through the valve. The water source was limited and could not maintain maximum flow throughout valve closure. Actual test measurements are proprietary and only the following "bottom-line" result is public. The valve failed to operate with the recommended operator torque settings supplied by the valve and operator manufacturers.

6. CONCLUSIONS

The valves that must be qualified to resolve GI–87 are pressure–seal, cast steel, gate valves in the 3 to 10 inch range and 600 and 900 lb. class. The most common manufacturer is Anchor/Darling Valve Company. Valve operators in use on these valves are electric motor driven (AC and DC) operators, manufactured by Limitorque Corporation. Typical system design conditions average 1250 psi and 575°F.

The mitigation of high energy pipe breaks are within the design basis for the above valves. Utilities typically purchase motor–operated valves which are certified under the manufacturer's Quality Assurance program to meet the design requirements established by the plant designer. Their method for establishing the qualification of the valve assemblies is to confirm that the certified performance of the motor–operated valve meets the design requirements of the system.

The same equation for sizing operators is used by most of the valve and operator manufacturers. This equation is simply the sum of three terms, the disc drag due to differential pressure load, the stem end pressure load, and the packing drag load. Flow through the valve is typically not factored into these equations. The equations depend heavily on the value used for the disc friction factor, which varies with vendor. Typical values are 0.2 and 0.3. This is inconsistent with recent test data, where disc friction factors ranged from 0.1 to 1.0.

The study of vendor methodology has identified several important parameters to be considered in the prediction of valve stem thrust loads. The specific relationship between these parameters and the stem thrust are not well understood. Differences of opinion exist in the following areas:

1. The effects of high mass flow on valve closure loads.

2. The ability to bound closing loads with substitute tests where the valve is opened starting at full differential pressure.

3. The correct disc friction factor for gate valves as a function of the other valve and operator parameters.

4. The effects of valve cycling on stem loads.

Very few tests under actual high energy pipe break conditions have been performed by utilities or valve and operator manufacturers. Only one vendor has blowdown isolation test experience, the others quote past experience in the commercial power industry to justify their methods. Operability of the valve assemblies is demonstrated using a substitute test where the valve is opened against full differential pressure. No data was found supporting the presumption that opening load with full differential pressure will bound the closing load at full system pressure and mass flow.

The few flow interruption tests that have been conducted, although not specifically designed to measure these phenomena, have identified the following general trends.

1. Repeated cycling tends to increase the valve thrust required to operate the valve.

2. The industry standard 0.3 disc friction coefficient is not conservative for all cases and may vary significantly from this nominal value. Coefficients have been measured from 0.1 to 1.0.

3. Mass flow/momentum could have a significant effect on valve stem thrust loads.

4. Increased temperature causes a significant increase in the required valve operating thrust.

The qualification of the isolation valves in the HPCI and RCIC steamlines and the RWCU suction line to close under high energy pipe break conditions is questionable. Evidence exists that, for some manufacturers, the actual stem load required to close the valve is quite different from the calculated stem load. Valves have failed to fully close in test programs where the valve assemblies were specifically designed for the test conditions using present qualification methods.

7. RECOMMENDATIONS

The review of test data and qualification techniques has provided information suggesting deficiencies in current closure load prediction and qualification practices. Further work is recommended as described below.

Additional independent test data should be obtained to clearly quantify the influence of the various parameters on valve closure loads. Based on the testing reviewed in this report, more information is required to provide confidence in our ability to define a conservative value for the friction load on the disc. Specifically the effects of cycling, seat and disc material specification, and temperature on the friction load should be evaluated. In addition, the previous results indicating that the friction load is proportional to pressure drop and independent of flow rate should be confirmed. Test data should be obtained through two methods:

1. Evaluate existing data from test laboratories, vendors, and the open literature. Test reports have been identified that, although proprietary, are available for review on-site.

2. Generating data from new independent tests.

The new testing would be designed to confirm selected important results from utility- or vendor-sponsored tests and to address anticipated deficiencies (gaps) in the existing experimental results.

8. REFERENCES

1. NRC memorandum from Harold R. Denton, Director Office of Nuclear Reactor Regulation to Robert B. Minogue, Director Office of Nuclear Regulatory Research, "Licensing Need For Selected Tasks Under Mechanical Equipment Qualification Research Program (FIN A6322)," March 11, 1986.

2. F. J. Mollerus, et. al., *Qualification of Active Mechanical Equipment For Nuclear Plants*, EPRI NP-3877, March 1985.

3. USNRC IE Bulletin No. 85–03, "Motor–Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings."

4. Letter from L. G. Kunel, Nebraska Public Power District to Robert D. Martin, Region IV, USNRC, "Response to IE Bulletin No. 85–03, Cooper Nuclear Station, NRC Docket No. 50–298, DPR–46," May 15, 1986.

5. Letter from Murray R. Edelman, The Cleveland Electric Illuminating Company to James G. Keppler, Region III, USNRC, "Perry Nuclear Power Plant, Docket Nos. 50–400 and 50–401, IE Bulletin 85–03, Motor–Operated Valve Failures," May 14, 1986.

6. Letter from Corbin A. McNeill, Jr., Public Service Electric and Gas Company to Dr. Thomas E. Murley, Region I, USNRC, "Motor–Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings Salem and Hope Creek Generating Station Facility Operating Licenses DPR–70, DPR–75 and NPF–50," May 27, 1986.

7. Anchor/Darling Valve Company, "Valve Seminar."

8. General Physics Corporation, "Nuclear Valve and Valve Operators," 1983.

9. *Marshall Electric Motor Operated Valve (Block Valve) Interim Test Data Report*, EPRI NP–2514–LD, July 1982.

10. C. A. Seaquist, *EPRI PWR PORV Block Valve Test Program: Final Review Report*, EGG–ED–6336, September 1983.

11. Westinghouse Electro–Mechanical Division, *EPRI Summary Report: Westinghouse Gate Valve Closure Testing Program*, March 31, 1982.

APPENDIX A

IDENTIFICATION OF GI-87 VALVE ASSEMBLIES

APPENDIX A

IDENTIFICATION OF GI–87 VALVE ASSEMBLIES

Two general tasks were undertaken to assess the population of the containment isolation valves used in the BWR systems of interest under GI–87. First was a review of available information sources to identify the systems applicable to GI–87 and to determine the valve and actuator manufacturers, types, and sizes used in those systems. The second task was a survey of vendors to determine industry methods of qualification. These tasks are discussed in the following paragraphs.

A.1 Identification of Systems Covered by GI–87

Information obtained from Preliminary/Final Safety Analysis Reports (PSARs/FSARs) permitted the identification of those systems that penetrate containment and directly communicate with the reactor vessel or recirculation lines. Tables A–1, A–2, and A–3 list the systems that meet these criteria. Table A–1 shows the systems for the BWR–2s, Table A–2 covers the BWR–3s and BWR–4s, and Table A–3 covers BWR–5s and BWR–6s. The fifth column in each table lists the operational status of the valves in each of the systems. Since GI–87 is concerned with the capability of isolation following a line break, only those systems with valves normally open were chosen for further study. The lines with check valves to prevent flow out of the reactor vessel were not investigated. The Main Steam Lines are also not included under GI–87.

The systems chosen for further study are listed below with a brief description of the specific line under consideration.

1. Emergency Cooling System—steam leaving reactor (BWR–2 only)
2. HPCI—turbine steam supply (BWR–3&4 only)
3. RCIC/Isolation Condenser—turbine steam supply
4. RWCU/Cleanup—water leaving reactor.

Table A–4 lists BWR plants and the plant–specific systems covered in this study.

Figures A–1 through A–4 are typical schematic drawings of these systems showing connections to the primary system and valve location and status. Hollow valve symbols indicate that the valve is open during normal plant operation.

A.2 Valve Sizes and Design Conditions

The PSAR/FSAR system data contained limited information about the containment isolation valves and operators. Restricting the search to the four systems previously identified, 84 process lines were studied. Each line has two containment isolation valves. With the exception of two plants where both isolation valves are located outside of containment, one valve is inside containment and the other is outside containment. All PSARs/FSARs containing operator information identified the inside containment isolation valve as having an AC power source, while the outside containment isolation valve had a DC source. Gate valves were identified as the type of valve used in all but two plants where globe valves were used. Complete system descriptions were not provided in all FSARs; however, the information available was very consistent from plant to plant and variation in those plants without a detailed FSAR are expected to be minor. The following paragraphs discuss the results of this literature search for the four chosen BWR systems.

The Emergency Cooling system is used only on BWR–2s. The system consists of two lines penetrating containment, each with two isolation valves located outside of containment. The system design pressure and temperature are 1250 psi and 575°F respectively. The pipe size for this system was not identified.

The HPCI is a 10–inch system with design pressures and temperatures ranging from 1120 to 1250 psi and 558 to 575°F respectively. All valves identified are gate valves.

Table A–1. Systems for BWR–2

Line or System	Number of Lines	Connection	Valves per Line	Status (Normal Position)
Main Stream	2	RPV	2	Open
Main Stream				
Warm–up	2	RPV	1	Closed
Emergency Cooling Vents	2	RPV	2	Open
Feedwater	2	RPV	2	Open/Check
Emergency Cooling				
Steam Leaving Reactor	2	RPV	2	Open
Cond. Return to Reactor	2	RECIRC	2	Closed/Check
Reactor Cleanup				
Water Leaving Reactor	1	RECIRC	2	Open
Water Return to Reactor	1	RECIRC	2	Open/Check
Shutdown Cooling				
Water Leaving Reactor	1	RECIRC	2	Closed
Water Return to Reactor	1	RECIRC	2	Closed
Reactor Head Spray	1	RPV	2	Closed/Check
Liquid Poison	1	RPV	2	Check
Control Rod Drive Hyd.	1	RPV	2	Check
Core Spray	2	RPV	3	2–Open/Closed

Table A-2. Systems for BWRs-3 and -4

Line or System	Number of Lines	Connection	Valves per Line	Status (Normal Position)
Main Steam	4	RPV	2	Open
Main Steam Drain	1	RPV	2	Closed
Feedwater	1	RPV	2	Open/Check
Reactor Water Sample	1	RPV	2	Closed
Control Rod Drive Return	4	RPV	2	Check
RWCU/Cleanup				
Water Leaving Reactor	1	RECIRC	2	Open
Water Return to Reactor	1	RECIRC	2	Open/Check
RHR				
Shutdown Cooling				
Supply	1	RECIRC	2	Closed
Return	2	RECIRC	2	Closed
LPCI Return to Reactor	2	RECIRC	2	Closed/Check
Reactor Head Spray	1	RPV	2	Closed/Check
Standby Liquid Control	1	RECIRC	2	Check
IC/RCIC				
Steam Supply	1	RVP	2	Open
Cond. Return	1	RECIRC	2	Closed/Open
Core Spray	2	RPV	2	Closed/Check
HPCI Steam Supply	1	RPV	2	Open

Table A–3. Systems for BWRs-5 and -6

Line or System	Number of Lines	Connection	Valves per Line	Status (Normal Position)
Main Steam	4	RPV	2	Open
Main Steam Drain	1	RPV	2	Closed
Feedwater	1	RPV	2	Open/Check
Reactor Water Sample	1	RPV	2	Closed
Control Rod Drive Retrun	4	RPV	2	Check
RWCU/Cleanup				
Water Leaving Reactor	1	RECIRC	2	Open
Water Return to Reactor	1	RECIRC	2	Open/Check
RHR				
Shutdown Cooling				
Supply	1	RECIRC	2	Closed
Return	2	RECIRC	2	Closed/Check
LPCI Return to Reactor	2	RPV	2	Closed/Check
HPCS Return to Reactor	1	RPV	2	Closed/Check
Standby Liquid Control	1	RECIRC	2	Check
RCIC Steam Supply	1	RPV	2	Open
RCIC RPV Head Spray	1	RPV	2	Check
Core Spray	2	RPV	2	Closed/Check

Table A–4. BWR plant listing

Plant Name	BWR Class	Type Containment	IC or RCIC	FWCI, HPCI, or HPCS	Cleanup or RWCU	Emergency Cooling
Oyster Creek	2	Mark I	IC	FWCI	Cleanup	ECCS
Nine Mile Point 1	2	Mark I	IC	FWCI	Cleanup	ECCS
Dresden 2 and 3	3	Mark I	IC	HPCI	Cleanup	—
Millsone 1	3	Mark I	IC	FWCI	Cleanup	ECCS
Monticello	3	Mark I	RCIC	HPCI	Cleanup	—
Quad Cities 1 and 2	3	Mark I	RCIC	HPCI	RWCU	—
Pilgrim	3	Mark I	RCIC	HPCI	RWCU	—
Brown's Ferry 1, 2, and 3	4	Mark I	RCIC	HPCI	RWCU	—
Vermont Yankee	4	Mark I	RCIC	HPCI	RWCU	—
Duane Arnold	4	Mark I	RCIC	HPCI	RWCU	—
Peach Bottom 2 and 3	4	Mark I	RCIC	HPCI	RWCU	—
Cooper	4	Mark I	RCIC	HPCI	RWCU	—
Hatch 1 and 2	4	Mark I	RCIC	HPCI	RWCU	—
Brunswick 1 and 2	4	Mark I	RCIC	HPCI	RWCU	—
Fitzpatrick	4	Mark I	RCIC	HPCI	RWCU	—
Enrico Fermi 2	4	Mark I	RCIC	HPCI	RWCU	—
Hope Creek	4	Mark I	RCIC	HPCI	RWCU	—
Susquahanna 1 and 2	4	Mark II	RCIC	HPCI	RWCU	—

Table A–4. (continued)

Plant Name	BWR Class	Type Containment	IC or RCIC	FWCI, HPCI, or HPCS	Cleanup or RWCU	Emergency Cooling
Shoreham	4	Mark II	RCIC	HPCI	RWCU	—
Limerick 1 and 2	4	Mark II	RCIC	HPCI	RWCU	—
La Salle County 1 and 2	5	Mark II	RCIC	HPCS	RWCU	—
WNP 2	5	Mark II	RCIC	HPCS	RWCU	—
Nine Mile Point 2	5	Mark II	RCIC	HPCS	RWCU	—
Grand Gulf 1 and 2	6	Mark III	RCIC	HPCS	RWCU	—
Perry 1 and 2	6	Mark III	RCIC	HPCS	RWCU	—
River Bend 1	6	Mark III	RCIC	HPCS	RWCU	—
Clinton 1	6	Mark III	RCIC	HPCS	RWCU	—

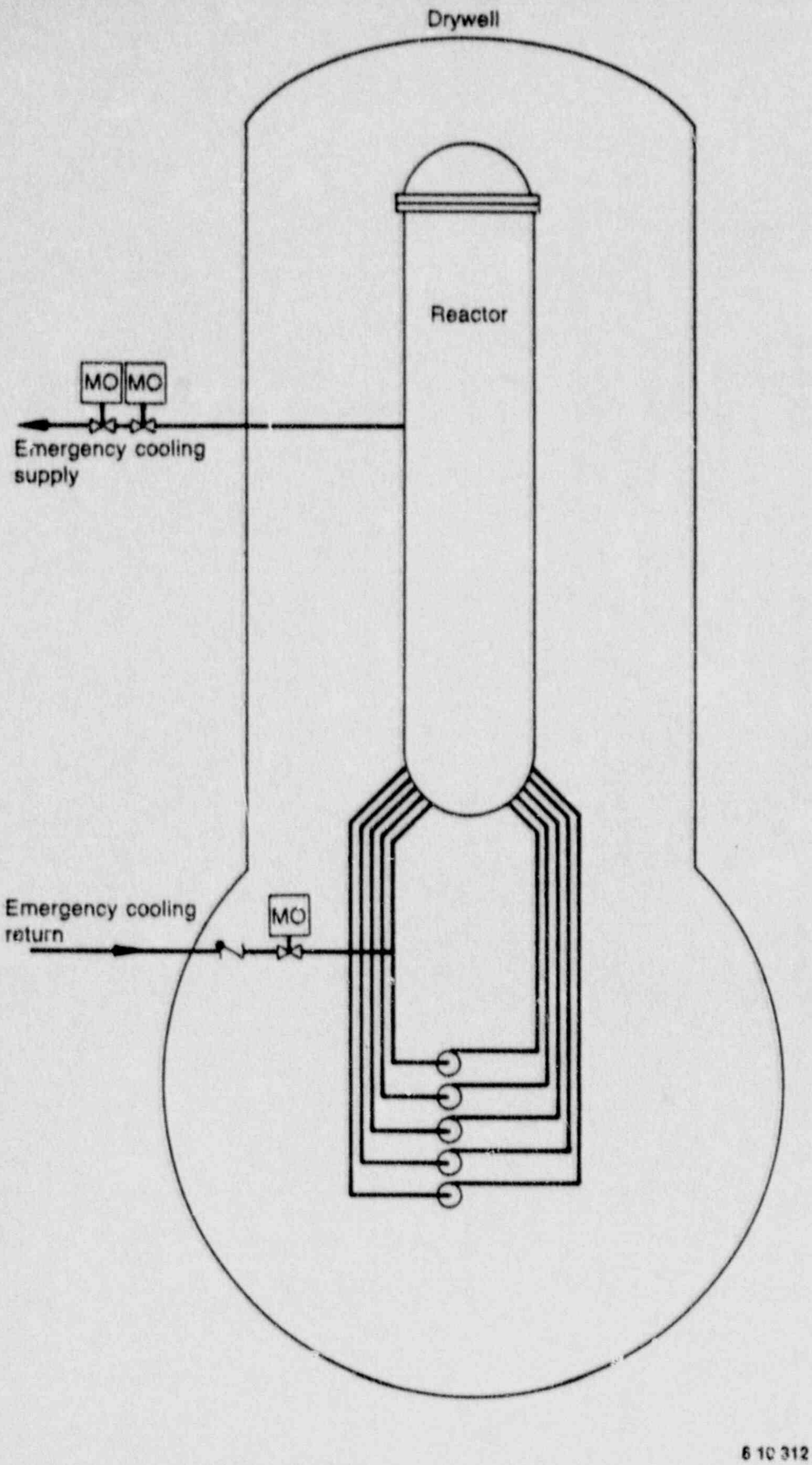

Figure A–1. Typical Emergency Cooling System.

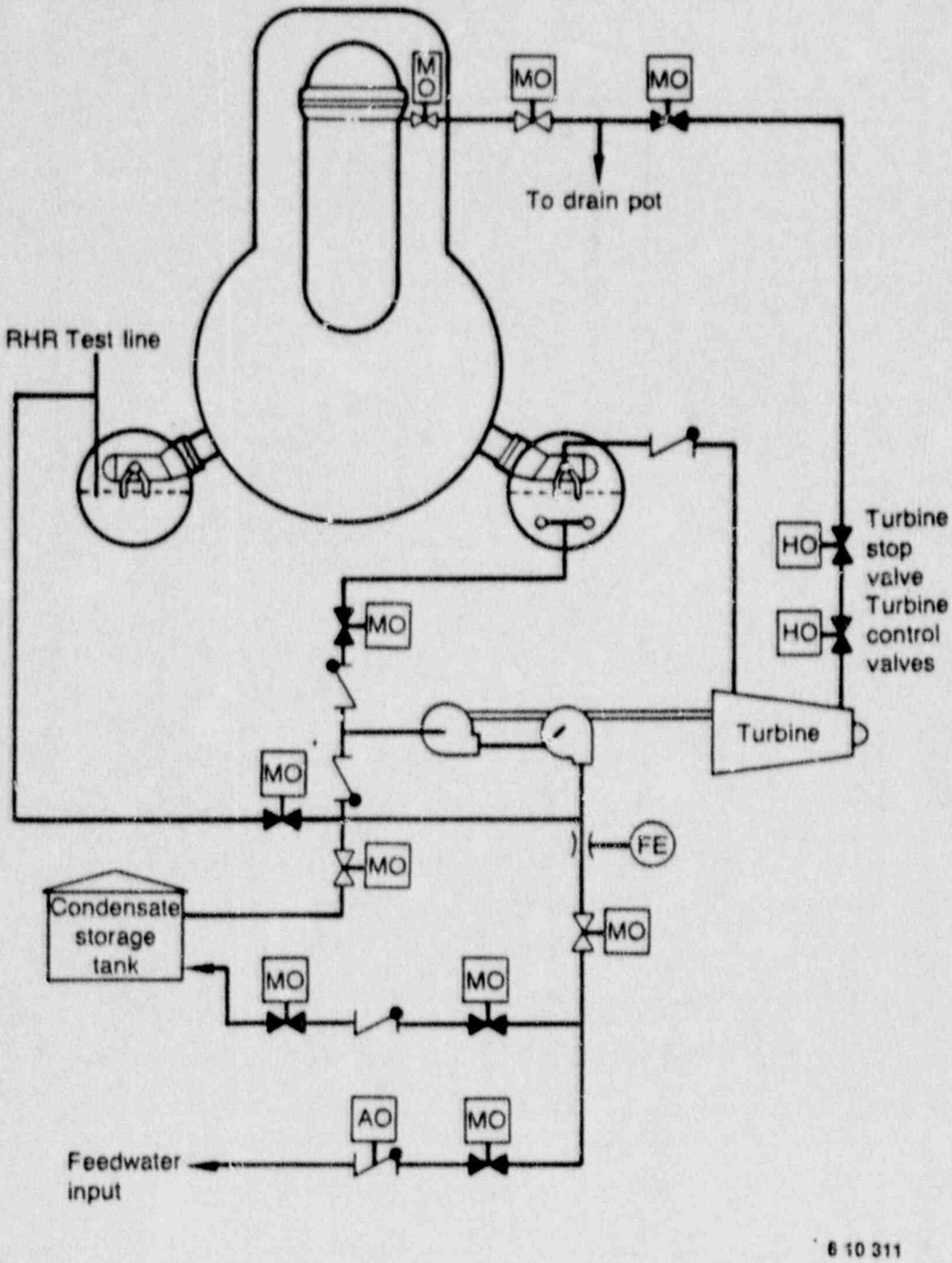

Figure A–2. Typical High Pressure Coolant Injection system.

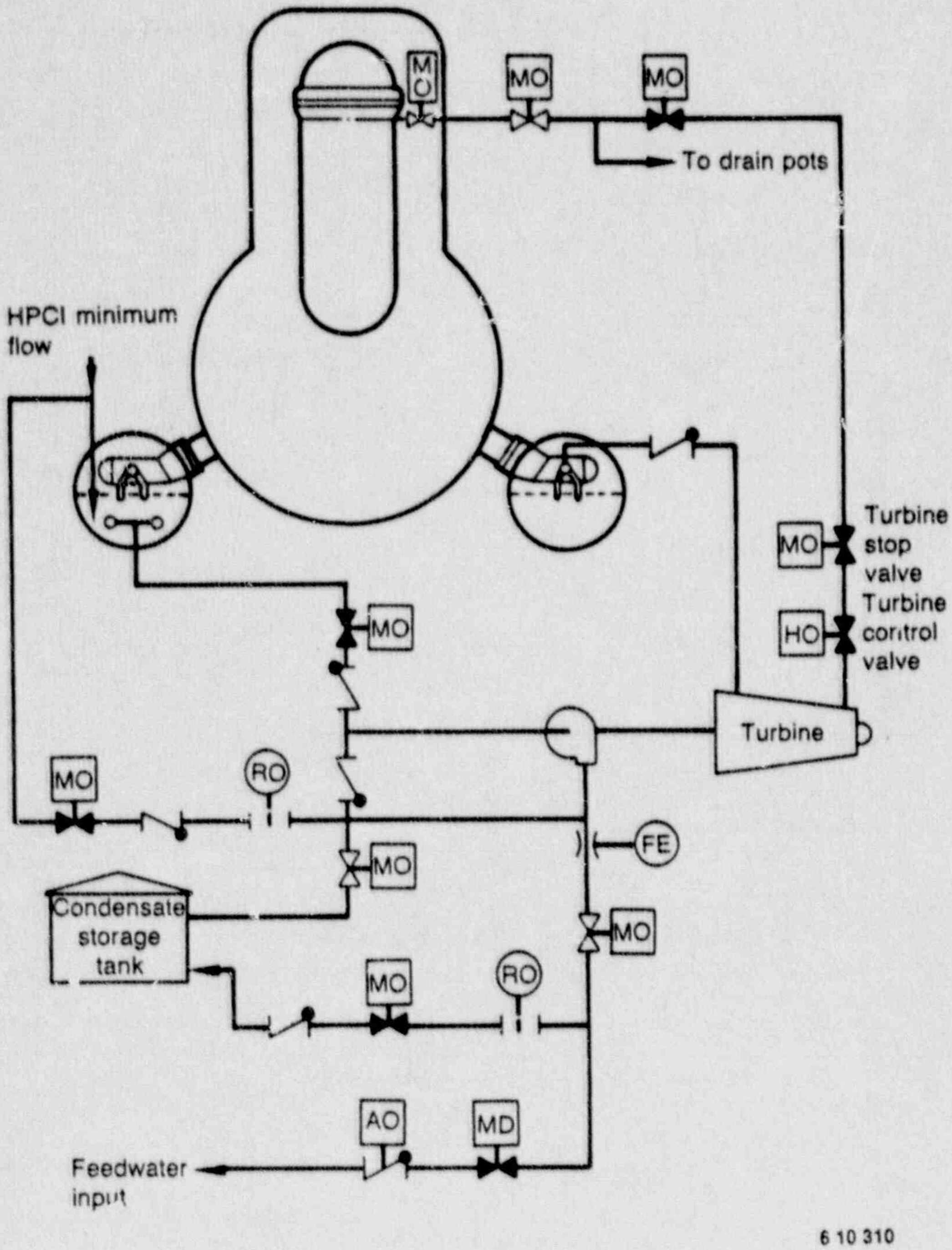

Figure A–3. Typical Reactor Core Isolation Cooling system.

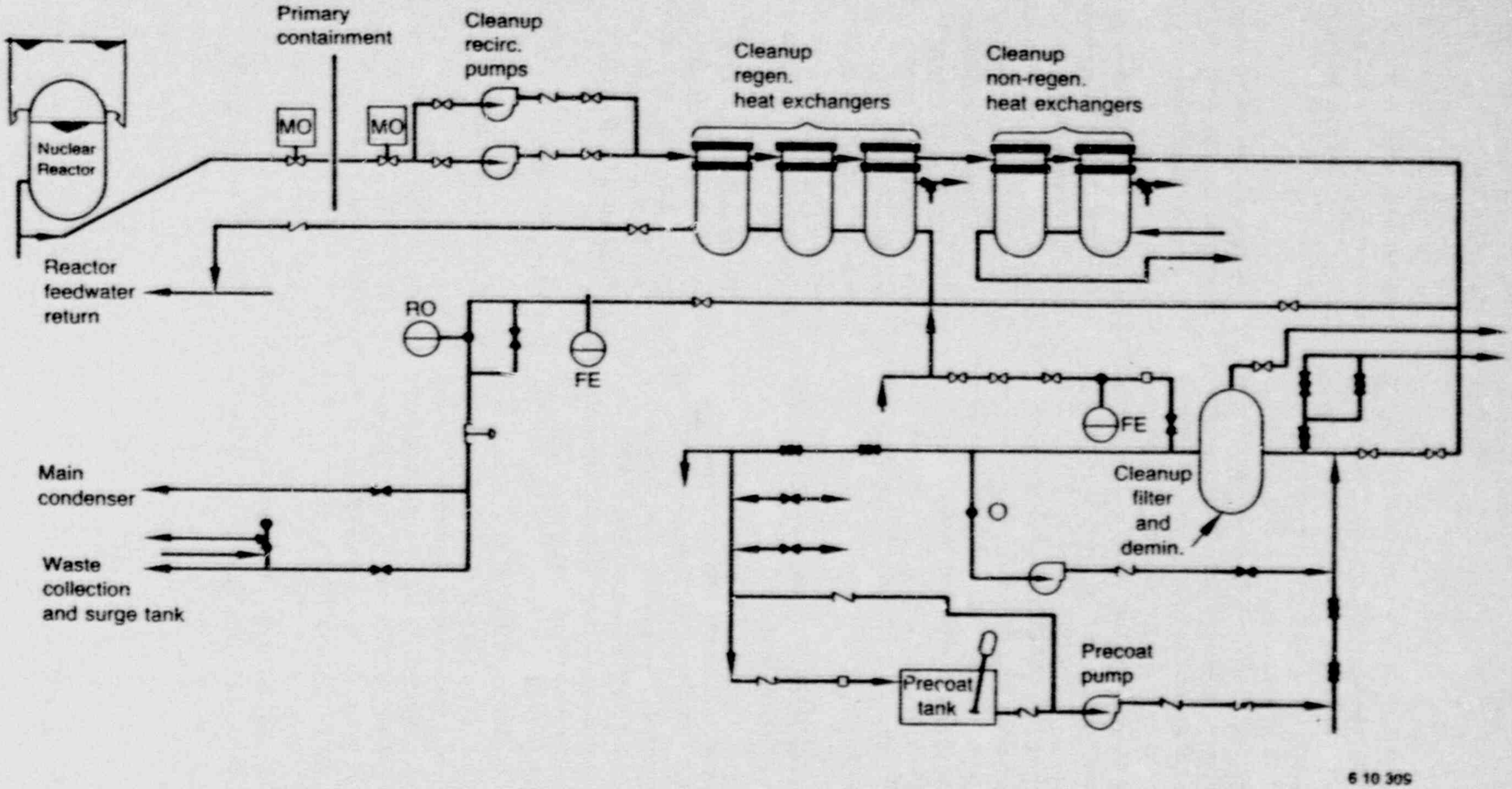

Figure A-4. Typical Reactor Water Cleanup system.

The RCIC and Isolation Condenser systems range in size from three to 14 inches with the majority being three and four inch lines and gate valves. Valves greater than four inches were identified in only three plants, one of which gave the valve configuration as two 10-inch gate valves with a 1-inch by-pass globe valve. The Isolation Condenser system identified in the BWR-2s and early BWR-3s contained the majority of the large (greater than 4 inches) valves. System design pressures and temperatures covered the same range as those for the HPCI system.

The majority of the RWCU systems include six-inch gate valves; three- and four-inch gate valves were identified in two plants each. The four oldest plants in the study use the Cleanup System which includes four-, six- and eight-inch valves. The design pressures and temperatures range from 1250 to 1450 psi and 564 to 575°F respectively.

Plant-specific system details are provided, as available from the FSARs, in Appendix B.

The Institute of Nuclear Power Operations (INPO) Nuclear Plant Reliability Data System (NPRDS) was used to determine specific valve assembly information. It provided the valve manufacturer, model number, type, size, maximum pressure and maximum temperature for the High Pressure Coolant Injection system (HPCI) and the Reactor Core Isolation Cooling system (RCIC). The NPRDS also provided valve operator manufacturer, model number, type, power source, maximum force, and maximum torque for these same two systems. The data base contained HPCI valve and operator data for 22 plants or 81% of the BWRs having that system; it contained RCIC valve and operator data for 24 plants or 67% of the plants having the RCIC system.

The most predominate valve manufacturer for both systems is the Anchor/Darling Valve Co. with 41% of the containment isolation valves. The other manufacturers are Borg-Warner (2%), Crane Co. (18%), William Powell Co. (11%), Velan Inc. (16%), and Walworth Co. (12%).

Limitorque Corporation manufactured 94% of the valve operators. The remaining 4 valve operators are identified as Philadelphia Gear Corporation operators.

Figures A-5 and A-6 show the distribution of HPCI and RCIC valve sizes among the various manufacturers. The HPCI systems (Figure A-5), with the exception of one plant, contain 10-inch gate valves exclusively while the RCIC systems contain 3-, 4-, 8-, and 10-inch valves. The containment isolation valves in the BWR-3s and BWR-4s are 3- and 4-inch gate valves, with the 3-inch valve being slightly more predominate. The RCIC lines were combined with the Residual Heat Removal (RHR) System in the BWR-5s and BWR-6s resulting in an increase in the pipe size to 8 and 10 inches. One plant has 8-inch valves while 3 plants have 10-inch valves.

The NPRDS data also contained the model number or vendor figure number for each valve. Vendor marketing literature and direct communication with vendor representatives identified the "typical" GI-87 containment isolation valve:

Type: Gate Valve

Size: 3 to 10 inches

Class: 600 and 900 lb

Body: Cast Steel

Bonnet: Pressure-Seal

Disc: Flexible Wedge.

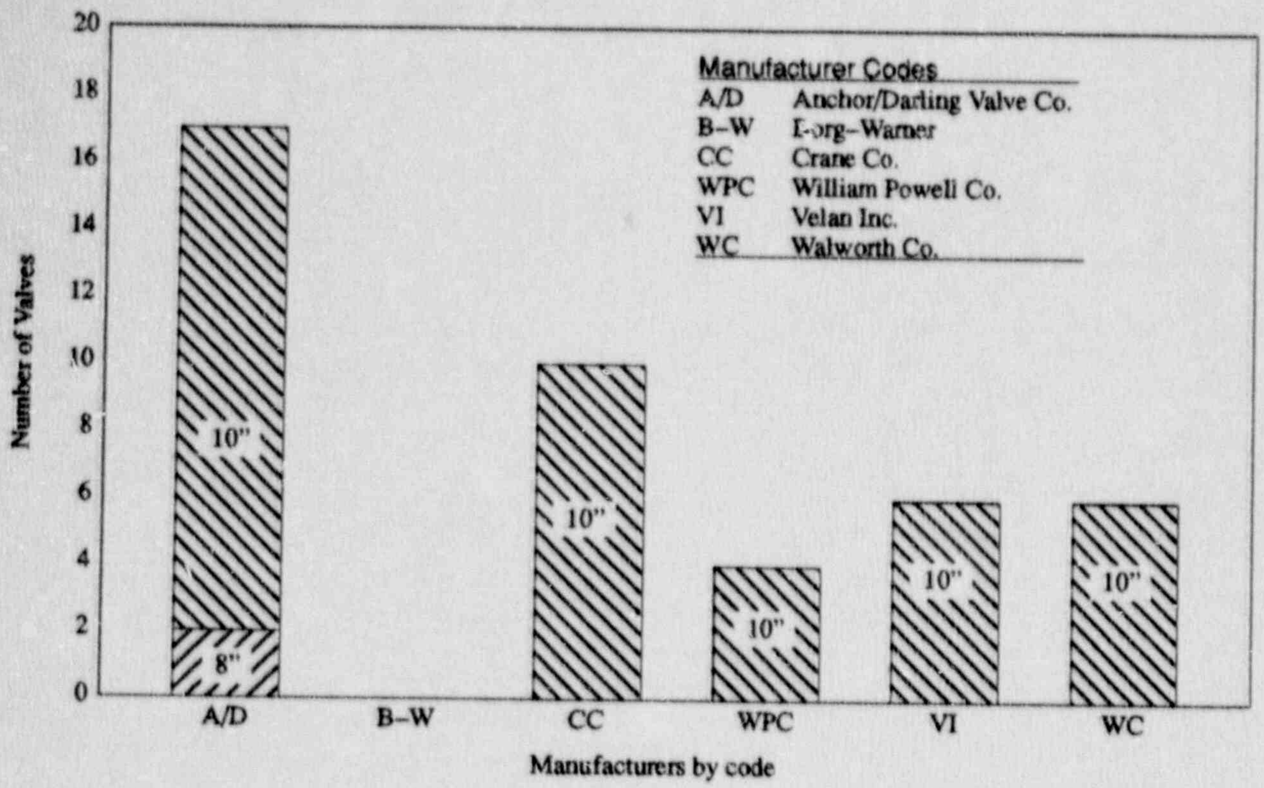

Figure A–5. HPCI isolation valve manufacturers by size.

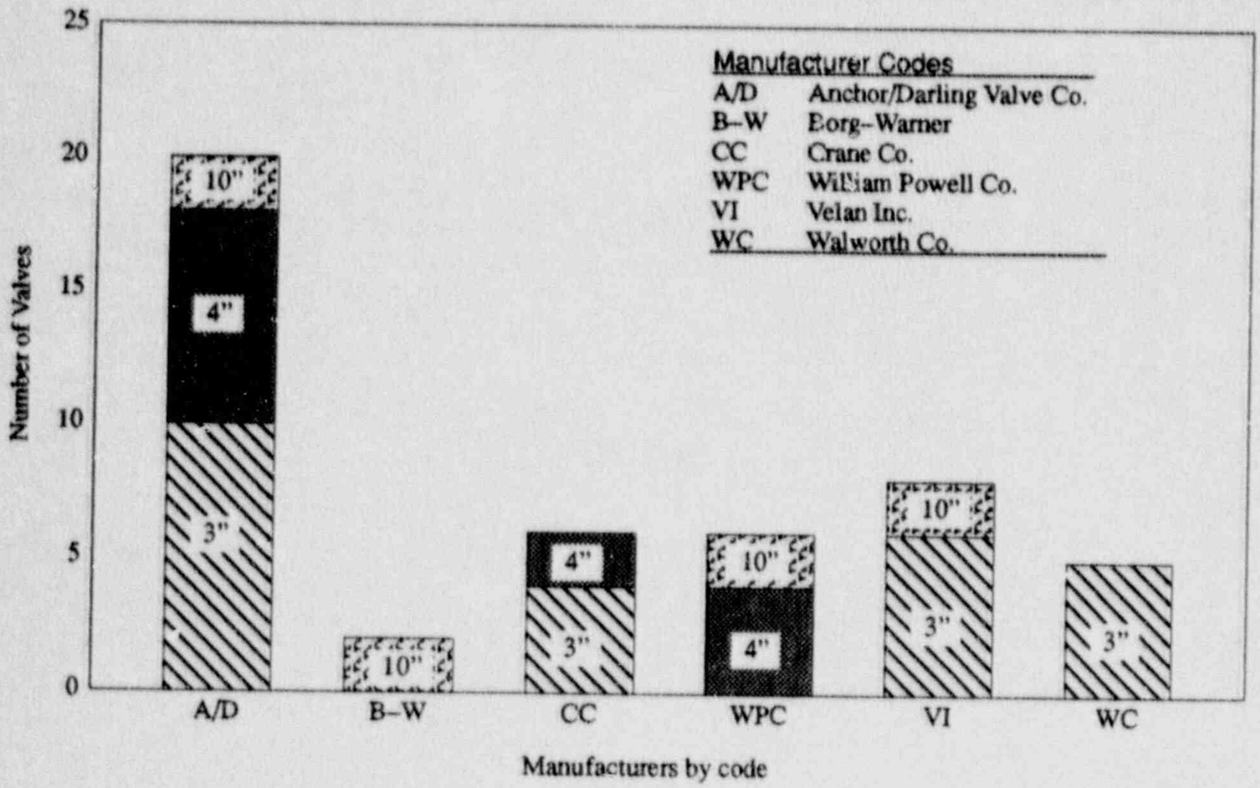

Figure A–6. RCIC isolation valve manufacturers by size.

APPENDIX B

PSAR/FSAR DATA

PSAR/FSAR DATA

Plant	BWR Class	Service	Pipe Size	Type	Status	Design PSIG	Design °F	Inside Valve ID	Outside Valve ID	Notes
Nine Mile PT.–1	2	Cleanup Supply	—	—	Open	1300	575	—	—	—
Oyster Creek	2	Cleanup Supply	6	—	—	1250	575	—	—	—
Dresden–2	3	Cleanup Supply	8	Gate	Open	1250	—	1201–1	1201–1	—
Dresden–3	3	Cleanup Supply	8	Gate	Open	1250	—	1201–1	1201–1	—
Monticello	3	Cleanup Supply	9	—	Open	—	—	MO–2397	MO2398	—
Nine Mile PT.–1	2	ECCS Steam Supply	—	—	Open	1250	575	—	—	a,b
Dresden–2	3	HPCI Steam Supply	10	Gate	Open	1125	558	2301–9	2301–5	—
Dresden–3	3	HPCI Steam Supply	10	Gate	Open	1125	558	2301–9	2301–5	—
Monticello	3	HPCI Steam Supply	10	—	Open	1125	558	MO–15	MO–16	—
Pilgrim–1	3	HPCI Steam Supply	—	Gate	Open	—	—	—	—	—
Quad Cities–1	3	HPCI Steam Supply	—	Gate	Open	—	—	2301–4	2301–5	—
Quad Cities–2	3	HPCI Steam Supply	—	Gate	Open	—	—	2301–4	2301–5	—
Browns Ferry–1	4	HPCI Steam Supply	10	—	Open	1120	—	—	—	—
Browns Ferry–2	4	HPCI Steam Supply	10	—	Open	1120	—	—	—	—

a. Two lines penetrate containment.

b. Both valves located outside containment.

PSAR/FSAR DATA (Continued)

Plant	BWR Class	Service	Pipe Size	Type	Status	Design PSIG	Design °F	Inside Valve ID	Outside Valve ID	Notes
Browns Ferry–3	4	HPCI Steam Supply	10	—	Open	1120	—	—	—	—
Brunswick–1	4	HPCI Steam Supply	—	—	Open	—	—	—	—	—
Brunswick–2	4	HPCI Steam Supply	—	—	Open	—	—	—	—	—
Cooper	4	HPCI Steam Supply	—	Gate	Open	—	—	—	—	—
Arnold	4	HPCI Steam Supply	—	—	—	—	—	—	—	—
Enrico Fermi–2	4	HPCI Steam Supply	10	Gate	Open	1250	575	E41F002	E41F003	—
Hatch–1	4	HPCI Steam Supply	—	Gate	Open	1250	575	—	—	—
Hatch–2	4	HPCI Steam Supply	—	Gate	Open	1250	575	—	—	—
Fitzpatrick	4	HPCI Steam Supply	10	—	Open	—	—	—	—	—
Limerick–1	4	HPCI Steam Supply	—	—	Open	—	—	—	—	—
Limerick–2	4	HPCI Steam Supply	—	—	Open	—	—	—	—	—
Peach Bottom–2	4	HPCI Steam Supply	10	Gate	open	—	—	—	—	—
Peach Bottom–3	4	HPCI Steam Supply	10	Gate	Open	—	—	—	—	—

Plant	BWR Class	Service	Pipe Size	Type	Status	Design PSIG	Design °F	Inside Valve ID	Outside Valve ID	Notes
Vermont Yankee	4	HPCI Steam Supply	10	—	Open	—	—	—	—	—
Oyster Creek	2	IC Return	10	—	Open	1250	575	—	—	—
Millstone–1	3	IC Return	—	—	Open	1250	575	—	—	—
Oyster Creek	2	IC Steam Supply	10	—	Open	1250	575	—	—	—
Dresden–2	3	IC Steam Supply	14	Gate	Open	—	—	1301–1	1301–1	—
Dresden–3	3	IC Steam Supply	14	Gate	Open	—	—	1301–1	1301–2	—
Millstone–1	3	IC Steam Supply	—	—	Open	1250	575	—	—	—
Monticello	3	RCIC Steam Supply	3	—	Open	1135	582	MO–2075	MO–2075	—
Pilgrim–1	3	RCIC Steam Supply	3	Gate	Open	1340	562	1301–16	1301–17	—
Quad Cities–1	3	RCIC Steam Supply	—	Gate	Open	1135	—	1301–16	1301–17	—
Quad Cities–2	3	RCIC Steam Supply	—	Gate	Open	1135	—	1301–16	1301–17	—
Browns Ferry–1	4	RCIC Steam Supply	3	—	Open	1146	562	—	—	—
Browns Ferry–2	4	RCIC Steam Supply	3	—	Open	1146	562	—	—	—
Browns Ferry–3	4	RCIC Steam Supply	3	—	Open	1146	562	—	—	—

Plant	BWR Class	Service	Pipe Size	Type	Status	Design PSIG	Design °F	Inside Valve ID	Outside Valve ID	Notes
Brunswick–1	4	RCIC Steam Supply	3	—	Open	1500	560	F007	F008	—
Brunswick–2	4	RCIC Steam Supply	3	—	Open	1500	560	F007	F008	—
Cooper	4	RCIC Steam Supply	3	Gate	Open	—	—	MO–15	MO–16	—
Arnold	4	RCIC Steam Supply	4	—	Open	—	—	MO–24	MD–24	—
Enrico Fermi–2	4	RCIC Steam Supply	4	Gate	Open	1250	575	E51F007	E51F008	—
Hatch–1	4	RCIC Steam Supply	3	Gate	Open	1250	575	F007	F008	—
Hatch–2	4	RCIC Steam Supply	3	Gate	Open	1250	575	F007	F008	—
Fitzpatrick	4	RCIC Steam Supply	3	—	Open	1250	575	MOV–15	MOV–16	—
Limerick–1	4	RCIC Steam Supply	3	—	Open	—	—	MO–15	MO–16	—
Limerick–2	4	RCIC Steam Supply	3	—	Open	—	—	MO–15	MO–16	—
Peach Bottom–2	4	RCIC Steam Supply	3	Gate	Open	1120	—	MO–15	MO–16	—
Peach Bottom–3	4	RCIC Steam Supply	3	Gate	Open	1120	—	MO–15	MO–16	—
Vermont Yankee	4	RCIC Steam Supply	3	—	Open	1250	575	—	—	a

a. Inside valve open/outside valve closed.

PSAR/FSAR DATA (Continued)

Plant	BWR Class	Service	Pipe Size	Type	Status	Design PSIG	Design °F	Inside Valve ID	Outside Valve ID	Notes
La Salle Co.–1	5	RCIC Steam Supply	10	Gate	Open	1250	575	E51F063	D51F064	a
La Salle Co.–1	5	RCIC Steam Supply	1	Globe	Close	1250	575	E51F076	—	b
La Salle Co.–2	5	RCIC Steam Supply	10	Gate	Open	1250	575	E51F063	E51F064	a
La Salle Co.–2	5	RCIC Steam Supply	1	Globe	Close	1250	575	E51F076	—	b
WNP–2	5	RCIC Steam Supply	3	Gate	Open	—	—	F007–1	F008	—
Clinton–1	6	RCIC Steam Supply	—	—	Open	1250	575	F063	F064	—
Perry–1	6	RCIC Steam Supply	—	Gate	Close	1250	575	F063	F064	—
Millstone–1	3	RWCU Supply	—	—	Open	1135	575	—	—	—
Pilgrim–1	3	RWCU Supply	6	Gate	Open	1340	575	1201–2	1201–5	—
Quad Cities–1	3	RWCU Supply	—	Gate	Open	1135	—	1201–2	1201–5	—
Quad Cities–2	3	RWCU Supply	—	Gate	Open	1135	—	1201–2	1201–5	—
Browns Ferry–1	4	RWCU Supply	6	—	Open	1146	575	—	—	—
Browns Ferry–2	4	RWCU Supply	6	—	Open	1146	575	—	—	—
Browns Ferry–3	4	RWCU Supply	6	—	Open	1146	—	—	—	—
Brunswick–1	4	RWCU Supply	6	—	Open	1500	564	F001	F004	—
Brunswick–2	4	RWCU Supply	6	—	Open	1500	564	F001	F004	—

a. Inside valve open/outside valve closed.

b. Bypass line.

PSAR/FSAR DATA (Continued)

Plant	BWR Class	Service	Pipe Size	Type	Status	Design PSIG	Design °F	Inside Valve ID	Outside Valve ID	Notes
Cooper	4	RWCU Supply	6	Gate	Open	—	575	MO–15	MO–16	—
Arnold	4	RWCU Supply	6	—	Open	—	564	MO–2700	MO–2701	—
Enrico Fermi–2	4	RWCU Supply	6	Gate	Open	1250	575	G33F001	G33F004	—
Hatch–1	4	RWCU Supply	6	Gate	Open	1250	575	F001	F004	—
Hatch–2	4	RWCU Supply	6	Gate	Open	1250	575	F001	F004	—
Fitzpatrick	4	RWCU Supply	6	—	Open	1250	575	MOV–15	MOV–18	—
Limerick–1	4	RWCU Supply	3	—	Open	—	564	MO–15	MO–18	—
Limerick–2	4	RWCU Supply	3	—	Open	—	564	MO–15	MO–18	—
Peach Bottom–2	4	RWCU Supply	6	Gate	Open	1120	575	MO–15	MO–16	—
Peach Bottom–3	4	RWCU Supply	6	Gate	Open	1120	575	MO–15	MO–16	—
Vermont Yankee	4	RWCU Supply	4	—	Open	1250	—	—	—	—
La Salle Co.–1	5	RWCU Supply	6	Gate	Open	1250	—	G33F001	G33F004	—
La Salle Co.–2	5	RWCU Supply	6	Gate	Open	1250	—	G33F011	G33F004	—
WNP–2	5	RWCU Supply	6	Gate	Open	—	—	F001	F004	—
Clinton–1	6	RWCU Supply	—	—	Open	1250	575	F001	F004	—
Perry–1	6	RWCU Supply	—	Gate	—	1250	575	F001	F004	—

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CR-5406
EGG-2569
Vol. 3

2. TITLE AND SUBTITLE

BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test

Review of Issues Associated with BWR Containment Isolation Valve Closure

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	1989

4. FIN OR GRANT NUMBER

A6857 and B5529

5. AUTHOR(S)

Kevin G. DeWall

Robert Steele, Jr.

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION – NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
P.O. Box 1625
Idaho Falls, ID 83415

9. SPONSORING ORGANIZATION – NAME AND ADDRESS (If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)

Division of Engineering and Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report presents the measured data and the analyses performed to date on the full-scale high-energy qualification and flow interruption gate valve testing to develop technical insights for the United States Nuclear Regulatory Commission (USNRC) effort regarding Generic Issue 87 (GI-87). The research was sponsored by the USNRC* and conducted by researchers from the Idaho National Engineering Laboratory. We tested two 6-in., 900-lb class valve assemblies, which represent a significant percentage of the reactor water cleanup isolation valves installed in plant applications. These valves were modified before testing by adding a high temperature load cell in the valve stems, which allowed the direct measurement of valve stem thrust during both opening and closing valve cycles. Instrumentation installed in the flow loop and on the valve assemblies measured the important valve and system test responses. Additionally, during the test program, all of the currently popular motor operated valve diagnostic test systems monitored the performance of the valves. Initially the valves were subjected to the hydraulic and leakage qualification tests defined in ANSI B16.41 and then to flow interruption and reopening valve tests at boiling water reactor primary system water temperature and pressure conditions with downstream line break flows. For the two valves tested, results show that (a) the disc factor used in current industry motor operator sizing equations underpredicts actual valve thrust requirements at all high temperature loadings, and for one valve design the equations may require an additional term to account for nonlinear performance, (b) the thrusts required to close the valves were sensitive to the fluid temperature, and (c) the results of testing at lower pressures, temperatures, and flows cannot be extrapolated to design basis pressures, temperatures, and flows for valve designs that have not exhibited linear performance behavior during design basis prototypical testing.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

valve testing
BWR reactor water cleanup
Generic Issue 87 (GI-87)

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)
Unclassified

(This Report)
Unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

SPECIAL FOURTH-CLASS RATE
POSTAGE & FEES PAID
USNRC
PERMIT No. G-67

120555139531 1 1AN1F1
US NRC-OADM
DIV FOIA & PUBLICATIONS SVCS
TPS PDR-NUREG
P-223
WASHINGTON DC 20555

NUREG/CR-5406, Vol. 3

BWR REACTOR WATER CLEANUP SYSTEM FLEXIBLE WEDGE GATE ISOLATION VALVE QUALIFICATION AND HIGH ENERGY FLOW INTERRUPTION TEST

OCTOBER 1989