Check Valve Hinge and Disc Assembly Discovered Unassembled

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Background

On June 21, 2000, at the Palisades Nuclear Power Plant, the High Safety Injection Pump P-66A failed to achieve its required flow reference value. Through evaluation, it was determined that the cause of this condition was that piston check valve number CK-ES3340 located in the mini flow recirculation line was stuck in a mid-stroke position. Check valve CK-ES3340 has a safety function in both the open and closed positions.

Once the cause was determined, a decision was made to designate swing check valve number CK-ES3332 to provide the safety functions of open and closed. This determination was based on past Inservice Testing (IST) and non-intrusive inspections of CK-ES3332 in both the open and closed direction.

In September 2000, radiography of the valve CK-ES3332 revealed that the valve internals were not attached. The plant shut down and reviewed all of the safety-related check valves. This investigation verified that all the check valves had been tested such that positive indication was provided for their operational readiness. This was done by verifying that the test methods, had without a doubt, proven that the valve obturator is intact and working correctly. If the testing method could not provide positive proof, then new testing was performed to provide this information. In almost all cases each test was backed up by multiple testing methods.

In 2000, the NRC issued NRC Information Notice (IN) 2000-21, "Detached Check Valve is Not Detected by Use of Acoustic and Magnetic Nonintrusive Test Techniques." In the summer of 2001, the Nuclear Industry Check Valve Group (NIC) provided the industry with guidance on this issue by developing an industry response to the IN. This response was sent out to all Vice Presidents and Managers at each nuclear site in the U.S.

Event Description

On September 5, 2000, at 1820 hours, a radiography of check valve CK-ES3332 in the train "A" common minimum flow recirculation line from high pressure safety injection (HPSI) pump P-66A and low pressure safety injection (LPSI) pump P-67A revealed that the check valve's disc/arm assembly was detached from the hinge pin and was located in the bottom of the check valve body.

Check valve CK-ES3332 was declared inoperable and technical specification 3.0.3 was entered based upon the potential for loose parts to affect additional components in the emergency core cooling system (ECCS).

The plant was shutdown and depressurized to 250 psi to effect repair—this isolated the mini flow lines to all ECCS pumps. Shutdown cooling was established with LPSI pumps in their shutdown cooling (SDC) mode with flow through the PCS allowing for their mini flow.

How Did We Get Here?

In June 2000, a problem with reduced HPSI pump P-66A recirculation flow had focused attention on various check valves in the train "A" common minimum flow recirculation line from HPSI pump P-66A and LPSI pump P-67A. CK-ES3340 was determined to be stuck in the mid-position. Note: CK-ES3332 is a significant valve due to its position in the common flow path for recirculation flow from the right channel engineered safeguards pumps, which include P-66A (HPSI), P-67A (LPSI), and P-54A (Containment Spray).

CK-ES3340 Nonintrusive Testing (NIT) Inconclusive

The quarterly P-66-A HPSI pump test indicated reduced flow and the acoustic NIT of CK-ES3340 was determined to be inconclusive. Radiography of CK-ES3332 and CK-ES3340 was parallel path with acoustic analysis.

First - Action on CK-ES3340

Radiography of CK-ES3332 was attempted at that time to determine whether the valve was contributing to the reduced recirculation flow. The radiography was inconclusive due to inadequate radiation source strength used for the radiography. Subsequently, upstream check valve CK-ES3340 was radiographed and was found to be partially open, (reference OE 11349) which explained the P-66A recirculation flow reduction symptom and further radiography of CK-ES3332 was no longer considered immediately necessary.

CK-ES3340 - Inspection Results

A Second Problem Arises

The September 5, 2000, radiography of CK-ES3332 was initiated with the intent of increasing the knowledge of the condition of the valve, based on a minimum amount of past data for it. The valve was not being radiographed because it was suspected of being failed.

A review of maintenance history, industry operating experience, design and application data revealed no problems.

The radiograph of CK-ES3332 performed on September 5, 2000, revealed that the check valve's disc/arm assembly was detached from the hinge pin and was positioned in the bottom of the check valve body.

CK-ES3332 Event Information

The initial supposition for the apparent condition of CK-ES3332 was service induced failure. However, when CK-ES3332 was opened for inspection, it was discovered that the disc and hinge assembly, including the disk nut, disk washer and cotter pin, were completely intact, laying in the bottom of the valve body and exhibiting no indication of failure from service wear. Accordingly, it was determined that the disc/arm assembly had not been attached to the hinge pin. This condition has likely existed since original plant construction, dating back approximately 30 years.

Safety Significance

CK-ES3332 has a safety function in the open direction to pass adequate minimum flow for

HPSI Pump P-66A, LPSI Pump P-67A and CS Pump P-54A. Observation over many years of pump operation and routine surveillance has demonstrated that the as-found condition of CK-ES3332 was not restricting recirculation flow.

Normally, CK-ES3332 has no safety function in the closed direction due to additional upstream check valves CK-ES3340 and CK-ES3233 for HPSI Pump P-66A and LPSI Pump P-67A, respectively. The upstream check valves are normally relied upon for closure in order to prevent the potential over-pressurization of an idle pump's suction piping.

Consequences of Taking Credit for Closure of CK-ES3332

In the ten-day period between June 21, 2000, and July 2, 2000, CK-ES3332 was credited with the closed safety function when radiography identified that upstream HPSI check valve CK-ES3340 was stuck in a mid-open position and, therefore, unable to provide the closed safety function. Prior to crediting CK-ES3332 with the closed safety function, non-intrusive testing (acoustic and dc magnetic testing) was performed, resulting in "apparent" open and closed indications. Based upon the as-found condition of CK-ES3332, it is apparent that the open and closed indications were caused by the disc/arm assembly responding to changes in flow.

Inspection of CK-ES3332

Detailed visual inspection of the condition of all wear surfaces, conclusively determined that this valve had never been in service in a fully assembled configuration. This conclusion is based on the following six inspection facts:

1. No rotational indications on either the hinge pin or swing arm.
2. No seat contact marks other than the initial bluing marks.
3. Because of the shape of the swing arm casting, an interference was found to exist between the hinge arm and valve body. If the valve had previously seen any actual service, this interference would prevent the valve disc from swinging open greater than 45 degrees and would have no indication of any impacts on the hinge arm back stop or valve body.
4. No indication of any impacts on the hinge arm back stop or valve body.
5. No evidence of rotation between the disc and disc stud.
6. The cast side areas of the body hinge pin bosses were in the rough cast condition, no rub marks could be found as would be expected from the disc arm rubbing on this surface.

Inspection of CK-ES3331

CK-ES3331 was inspected by boroscope and found to be in excellent condition during the repair of CK-ES3332. The area where the tail-piece on the hinge arm contacts the body was examined.

It was clearly evident that they contact each other—meaning that an interference fit does not exist. It was baseline tested with the P-67B LPSI , using both dc magnetics and acoustic NIT methods.

Risk Impact of Event

The Palisades Probabilistic Safety Assessment (PSA) was evaluated for the risk impact due to CK-ES3332 being unable to provide the closed safety function. The only period during which the as-found condition of CK-ES3332 would have caused potential concerns was during the ten-day period when it was credited with a closed safety function. Since neither CK-ES3332 nor CK-ES3340 were capable of closure during this time period, an evaluation of possible operating and accident scenarios was performed to identify the maximum pressure that could be experienced in HPSI Pump P-66A suction piping for comparison to design pressure ratings. The section of piping between HPSI Pump P-66A, upstream check valve CK-ES3183 and upstream branch isolation valve CV-3071 was identified as having the potential to be pressurized beyond design pressure to a maximum of 1250 psi. While this section of piping is rated for 500 psi, and the aforementioned valves are rated for 300 psi, evaluation has concluded that the piping and valves would have maintained structural integrity under this increased pressure loading.

Actions Taken

CK-ES3332 was inspected and reassembled, restoring it to its intended condition. An interference fit between the hinge arm and the body had to be corrected to allow the disk to full open (some material was trimmed off the hinge arm).

A restart review of all IST Program check valves was performed to ensure that an adequate basis existed to conclude that each check valve is functioning properly. Where necessary, corroborating data was obtained via additional testing. No other anomalies or degraded conditions were identified from this effort.

Review of Previous Nonintrusive Testing for CK-ES3332

During two previous nonintrusive tests (11/97 and 6/00), the actual condition of CK-ES3332 was not ascertained. Though acoustic testing was performed, prior to discovering the disc laying on the bottom of the valve by radiograph testing (RT), results obtained from acoustic testing corresponded with the generically expected indications.

Lessons Learned

A review of industry experience for deficiencies in the application of non-intrusive testing (such as acoustic testing) was performed. Lessons learned from reviewing the search material, in addition to conclusions reached from this event, reinforce the need to use more than one confirmatory technique for valve condition when using nonintrusive techniques.

Another common theme noted is the need for acquiring "good" baseline measurements when using acoustic monitoring technology, i.e., the need to have reasonable assuredness of existing valve condition, that consistent test conditions are used, and that proper operation is established as part of the baselining process.

Some points to make:

The level of baseline testing for each type of check valve is not the same.

- A one piece piston check would not loose its disc into the system (generate loose parts), whereas a swing check valve could loose the disc nut, hinge pin, hinge arm, etc.
- Tilting disc checks generally do not have a hinge arm that could move and impact the backstop, if the disc was dislodged from the hinge pins, whereas if the disc separated from the hinge arm in a swing check, the hinge arm could still move and hit the backstop.

- Nonintrusives that are used should be selected to provide reasonable assurance of valve condition, such as when using acoustics, to collaborate impact data with backflow data (DP / leakage / flow), RT, UT, AC/DC Magnetics, or previous inspection results.

Is There More Information We Are Missing?

The Palisades event was a very unique situation in that the disc was unassembled from preservice days and that it had passed its operability flow test for 30 years. Having looked at the maintenance history (none found except an acoustic NIT), industry operating experience, design and application data, there were no problems expected. The acoustic data by itself did not lead one to believe otherwise.

Question—How many cases have actually been recorded where a single valve in a sample group degraded / acted drastically different from the group as a whole? To date we have not come across any. Normally, the condition of one valve in the group is representative of the whole. However, this was not the case for the CK-ES3332 failure at Palisades.

NRC Concerns

The NRC considers NIT acceptable for in-service testing of check valves provided that the method used is qualified. If the owner should use NIT, they need to establish a performance baseline in both directions when the check valve is in a known acceptable operating condition. A check valve's performance can then be assessed against this baseline. Both the NRC and industry have provided guidance on the use of NIT.

When using NIT, it is also important for the test conditions to be repeatable so that the test results can be reviewed with prior tests. NIT techniques need to be accurate and repeatable. When NIT is to be used, it needs to be verified that the method being used will determine the valve's function that is being detected. The qualification process may reveal that certain NIT techniques give inconclusive results for a particular application.

The NRC Information Notice (IN) 2000-21, "Detached Check Valve Disc Not Detected By Use of Acoustic and Magnetic Nonintrusive Test Techniques," concludes:

"If NIT techniques used to verify the opening or closing capability of safety-related check valves are not properly qualified and a baseline established for each individual valve when the valve is known to be operating acceptably, potentially inadequate valve performance may be undetectable in the analysis of NIT results."

An Industry Unified Response

As a result of this IN, the Nuclear Industry Check Valve Group (NIC) provided an industry response to this notice. A letter was sent from NIC to all Site Vice Presidents and Managers. The letter was developed and voted on by utility members that were present at the Summer 2001 Meeting.

In addition to this letter, NIC continues to move the industry forward on this and many other issues. Presently, NIC is conducting a Check Valve Performance Trending Initiative. A recommended practice by the NRC and INPO is to trend check valve conditions, so that maintenance is performed prior to failure.

This NIC test initiative addresses this industry need. The scope of work proposed is to conduct testing to evaluate the capabilities of various commercially available techniques and technologies to trend parameters that would reveal the internal condition of check valves. To effectively utilize these technologies, further verification of their capability to trend parameters in detecting check valve degradation is desired. The results of this initiative should allow utilities to demonstrate that monitored and trended parameters are repeatable, reliable and defensible. Effective trending is expected to result in substantial reductions in both operation and maintenance costs.

References

1. NRC Information Notice 2000-21, "Detached Check Valve Disc Not Detected By Use of Acoustic and Magnetic Nonintrusive Test Techniques."

2. NRC Temporary Instruction 2515/110 (Revision 1), Performance of Safety-Related Check Valves.

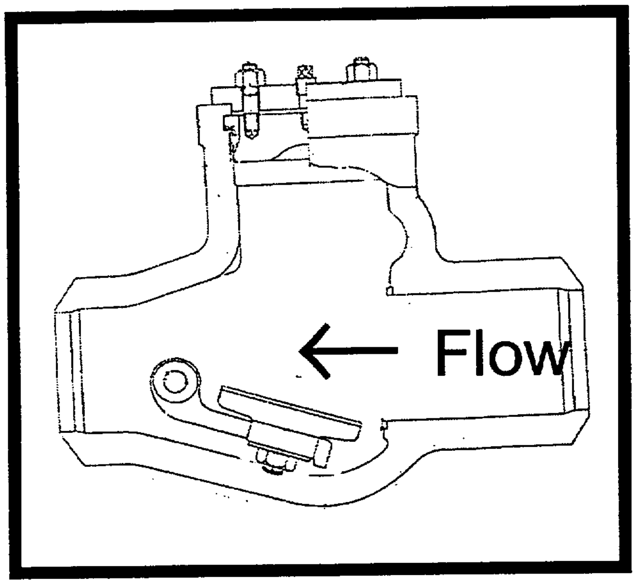
3. ASME Inquiry OMI-00-09, June 2001—This inquiry is applicable to OMa-1988, Part 10, paragraph 4.3.2.4, Valve Obturator Movement and Paragraph 3.3, Reference Values.

4. ASME Inquiry OMI-00-08, June 2001—This inquiry is applicable to OMa-1988, Part 10, Paragraph 4.3.2.2, Exercising Requirements; Paragraph 4.3.2.4, Valve Obturator Movement; and Paragraph 3.3, Reference Values.

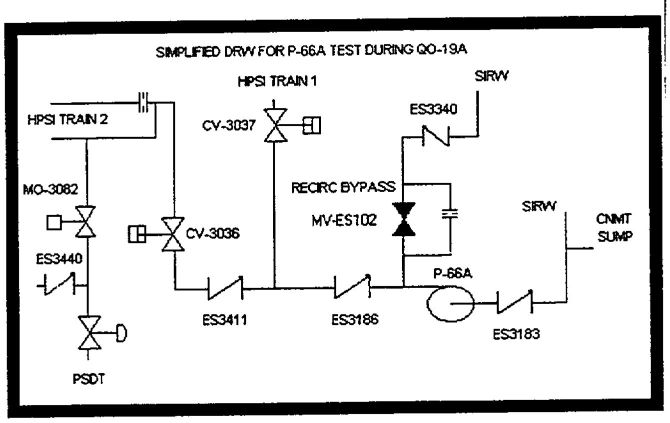
5. LER 50-225/00-04, "Discovery of Inoperable Check Valve Results in Plant Shutdown," October 4, 2000 (Accession No. 9810270327).

6. NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," April 1995.

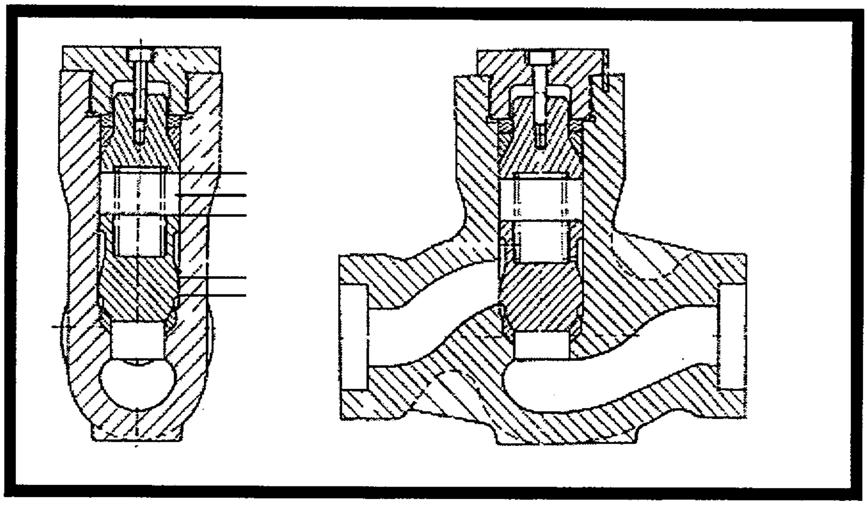
7. "Evaluation of Nonintrusive Diagnostic Technologies for Check Valves (NIC-01)," Volume 1, February 1991, transmitted by a letter dated February 20, 1992, to Francis Grubelich, NRC, from the Nuclear Industry Check Valve Group (Accession No. 9205280219).



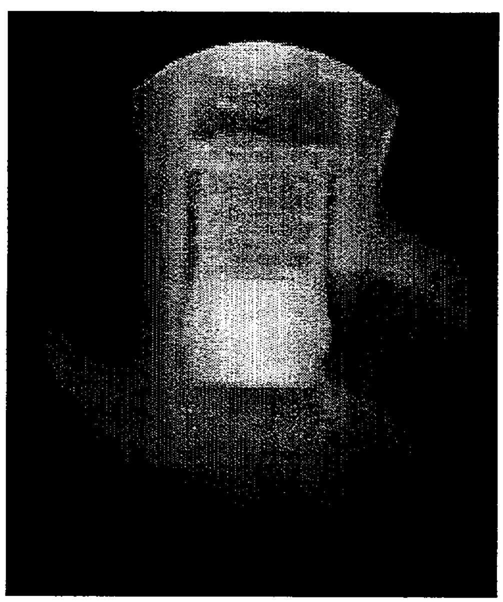
900# Swing Check Valve



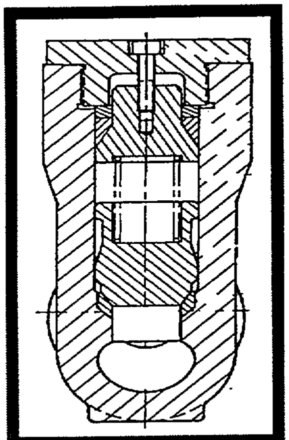
HPSI pump 66A test

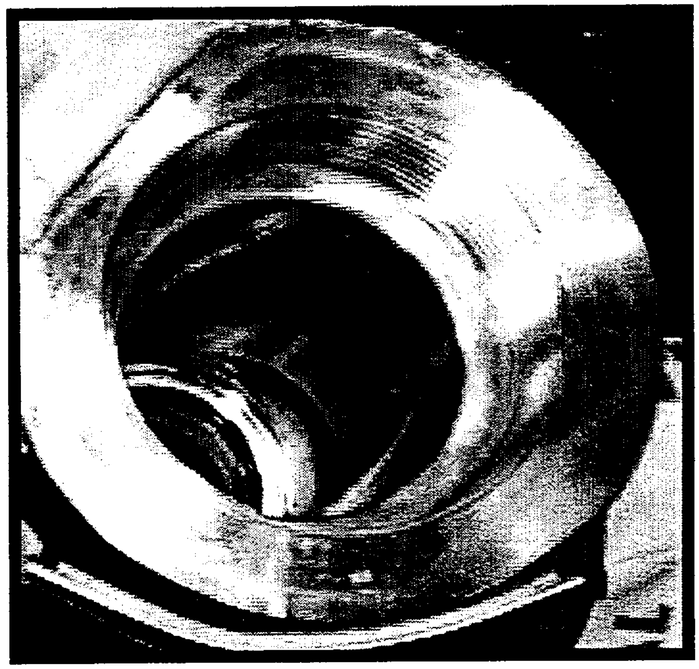


2-inch T-pattern Piston Check Valve

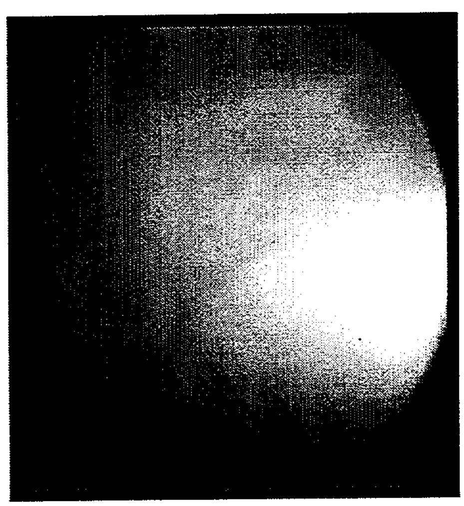


Radiograph of CK-ES3340 showing piston stuck open

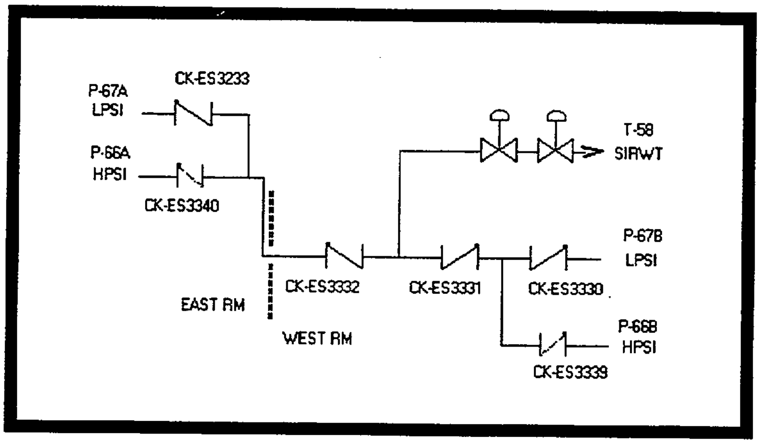




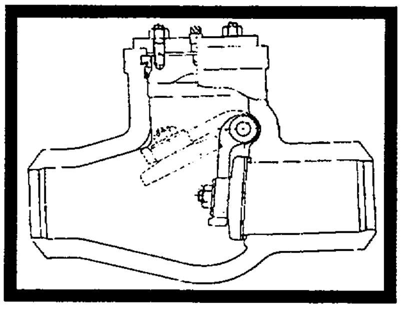
View looking into cylinder / body showing fretting area at outlet port.



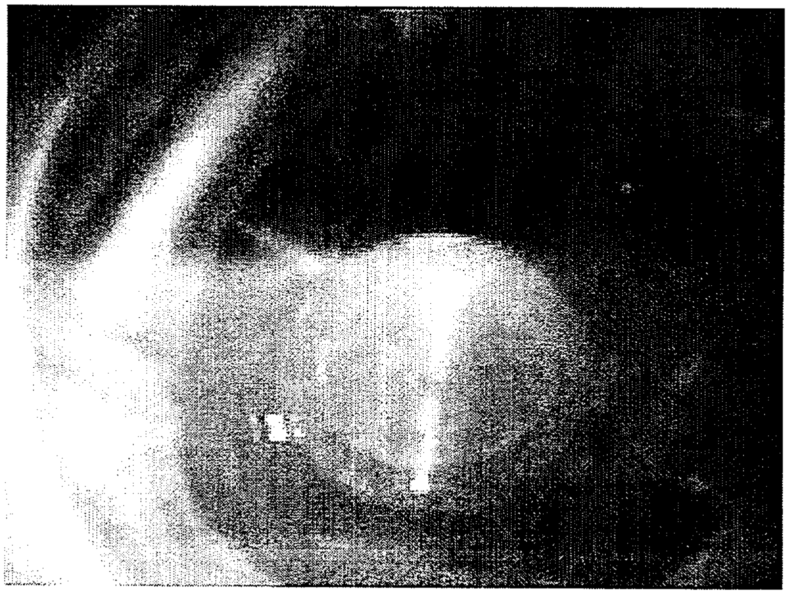
Radiograph of CK-ES3332 (side view) showing disk laying on bottom of valve. The seat side of the disk was facing up and the hinge arm was nearer the outlet port.



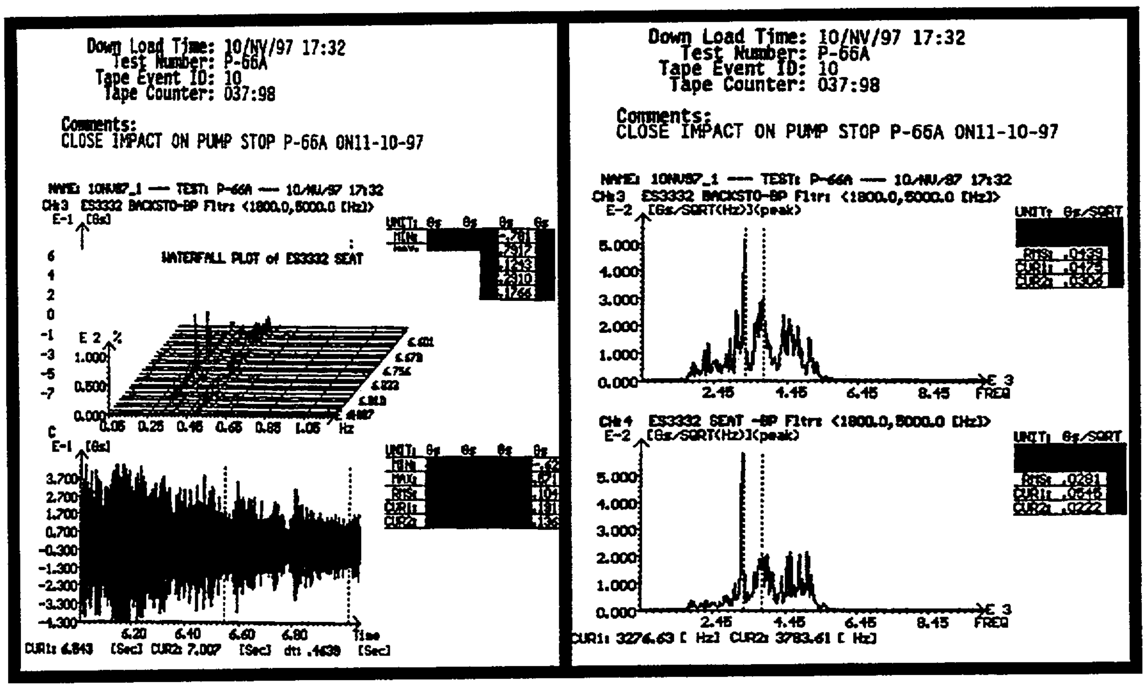
CK-ES3332 System Lineup



Swing Check Valve

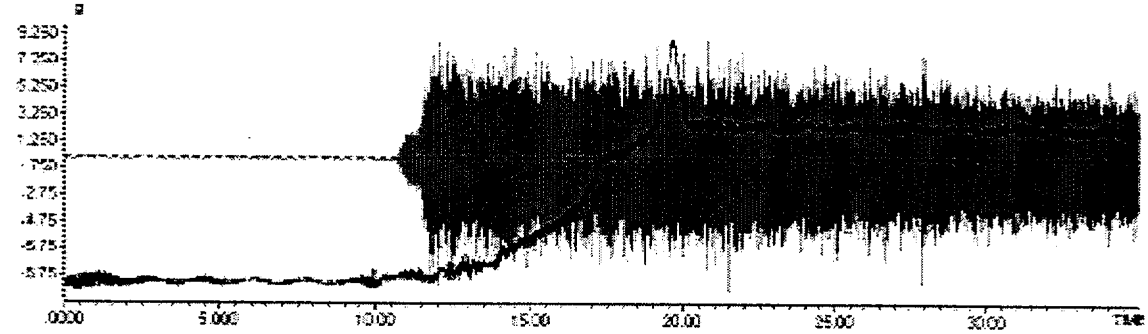


CK-ES3332 Internals

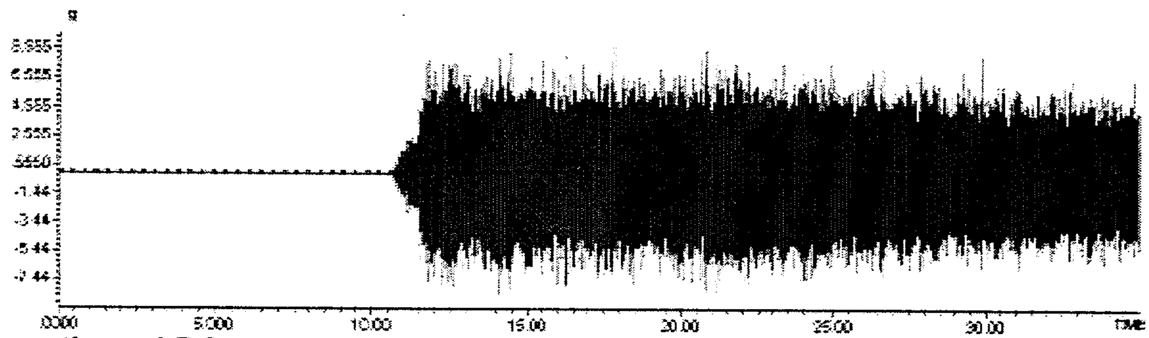


11/97 Test results showing close impact amplitude and frequency data.

CH -1: CK-ES3332 BACKSTOP w/ DC MAGNETIC OVERLAY - BP FILTER 1000 - 6500 HZ

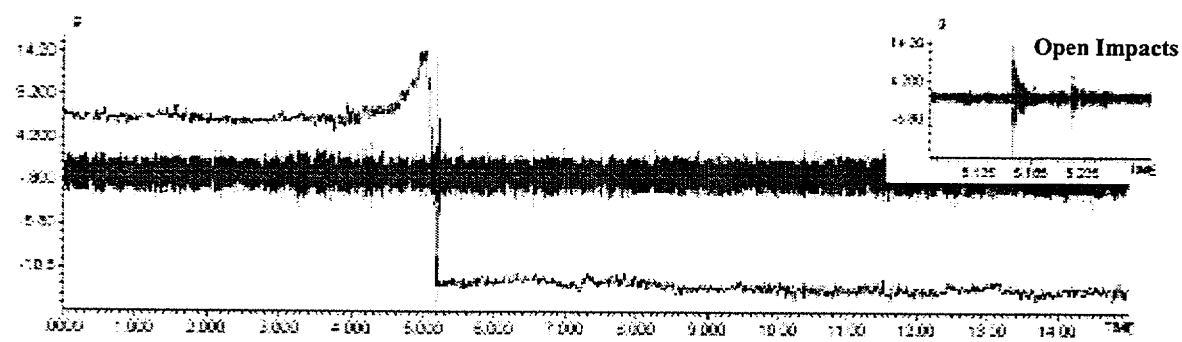


CH -2: CK-ES3332 SEAT - BP FILTER 1000 - 6500 HZ

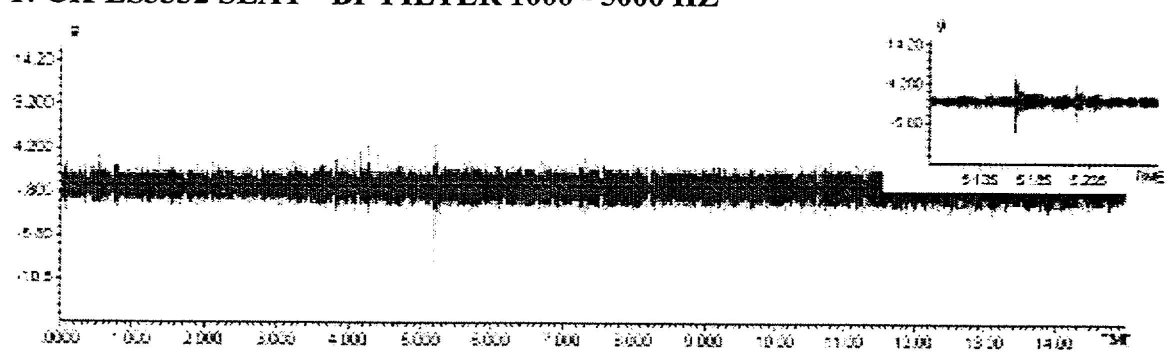


Acoustic and DC magnetic traces of CK-ES3332 open test (credit for operability - 6/21/00) using the P-66A, HPSI pump for flow. CK-ES3340 is stuck open at this time.

CH - 1: CK-ES3332 BACKSTOP w/ DC TRACE OVERLAY - BP FILTER 1000 - 5000 HZ

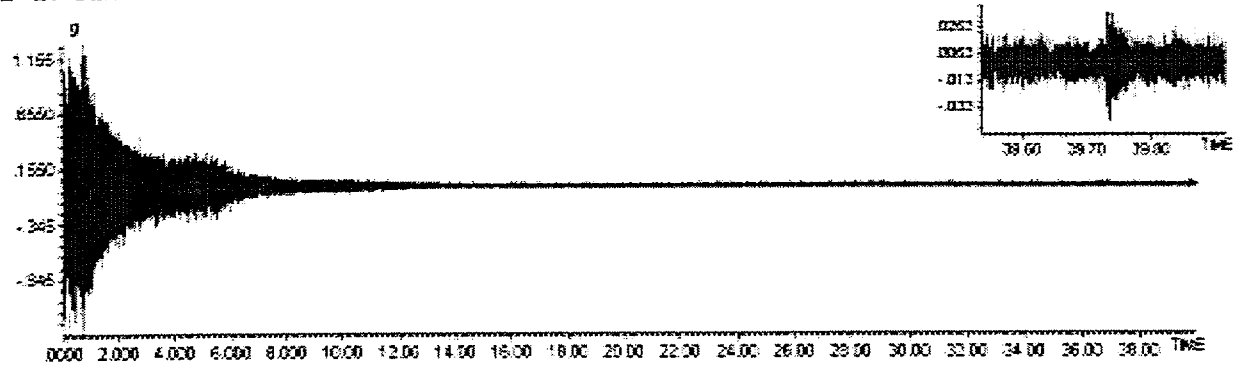


CH - 1: CK-ES3332 SEAT - BP FILTER 1000 - 5000 HZ

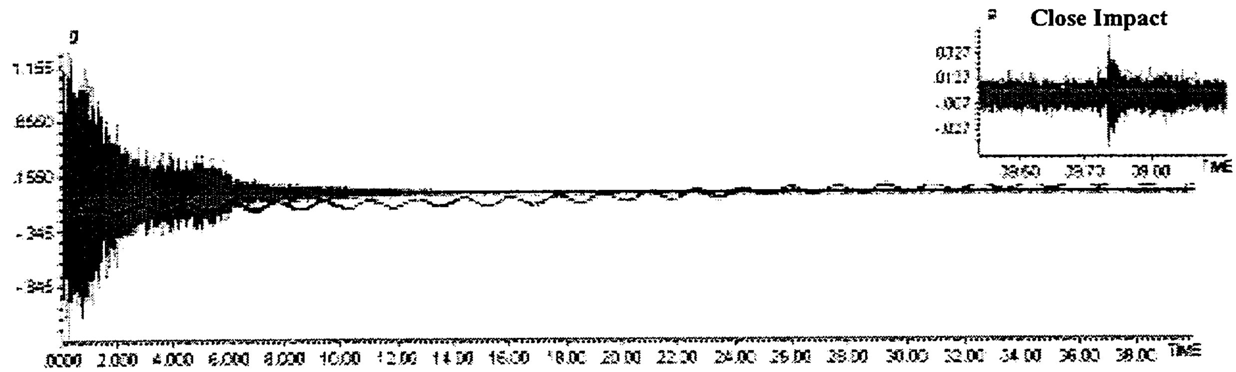


Acoustic and DC magnetic traces of CK-ES3332 open test (credit for operability) using the P-66A, HPSI pump for flow. CK-ES3340 is stuck open at this time.

CH -1: CK-ES3332 BACKSTOP - BP FILTER 1000 - 6500 HZ

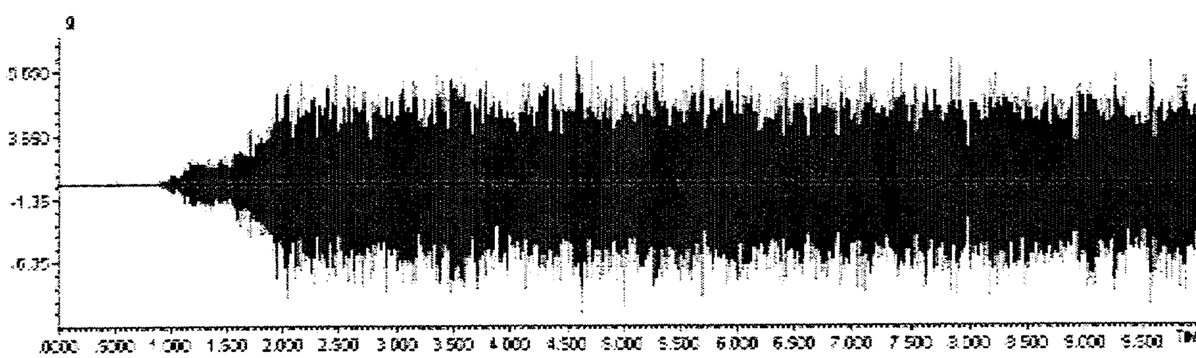


CH -2: CK-ES3332 SEAT - BP FILTER 1000 - 6500 HZ with DC TRACE OVERLAY

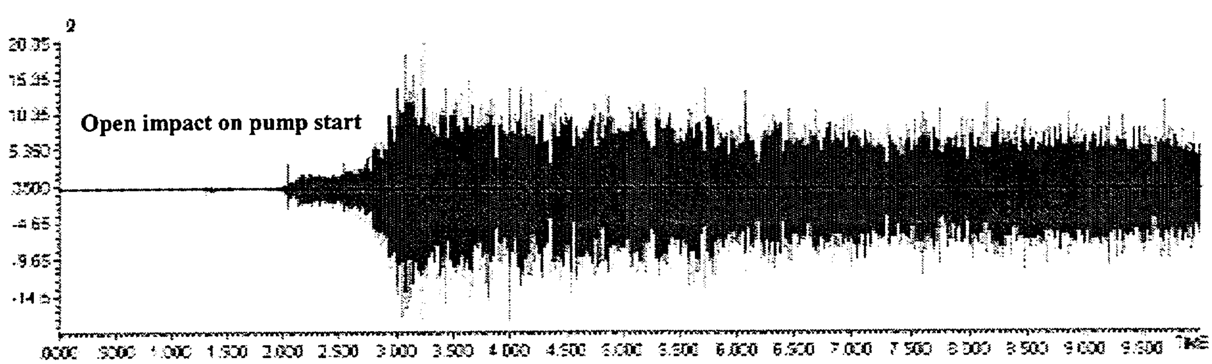


Acoustic and DC magnetic traces of CK-ES3332 close test (credit for operability) using the P-66A, HPSI pump for flow. CK-ES3340 is stuck open at this test.

CH -3: CK-ES3332 BACKSTOP - BP FILTER 1000 - 6000 HZ

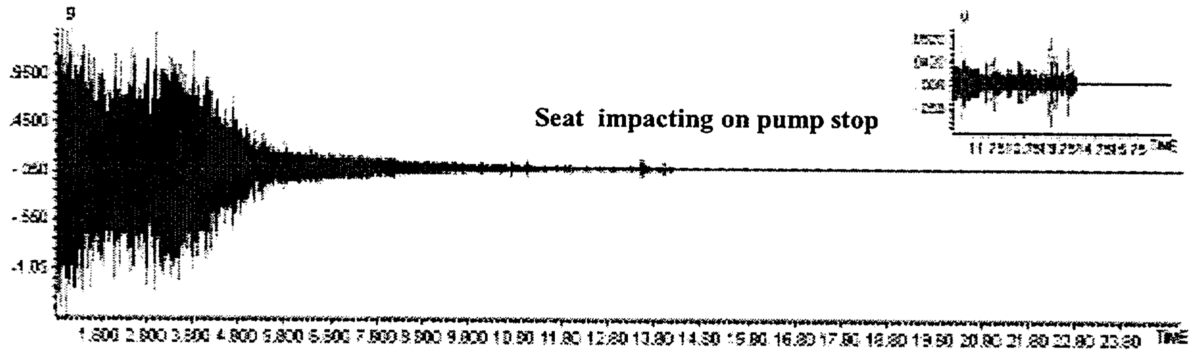


CH -1: CK-ES3331 BACKSTOP - BP FILTER 1000 - 6000 HZ

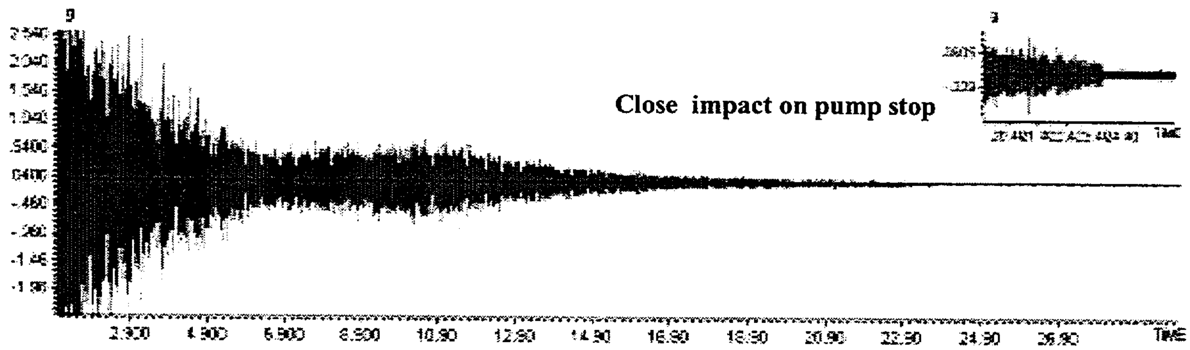


Open acoustic traces of CK-ES3332 and CK-ES3331 using the HPSI pumps for flow. The 11/97 DAT data was downloaded into a newer version of software.

CH -3: CK-ES3332 SEAT - BP FILTER 1000 - 6000 HZ

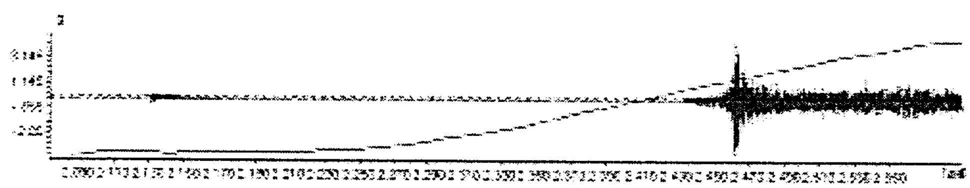


CH -1: CK-ES3331 SEAT - BP FILTER 1000 - 6000 HZ

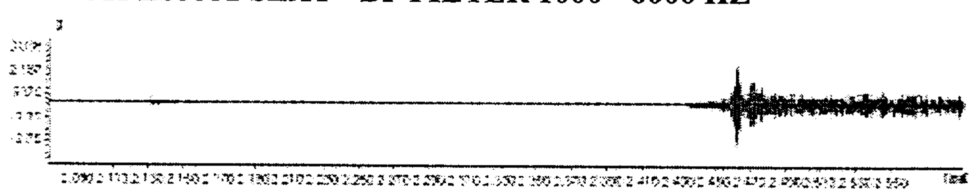


Close acoustic traces of CK-ES3332 and CK-ES3331 using the HPSI pumps for flow. The 11/97 DAT data was downloaded into a newer version of software.

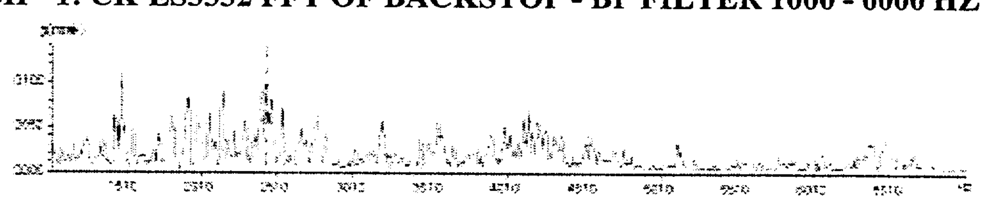
CH - 1: CK-ES3332 BACKSTOP - BP FILTER 1000 - 6000 HZ



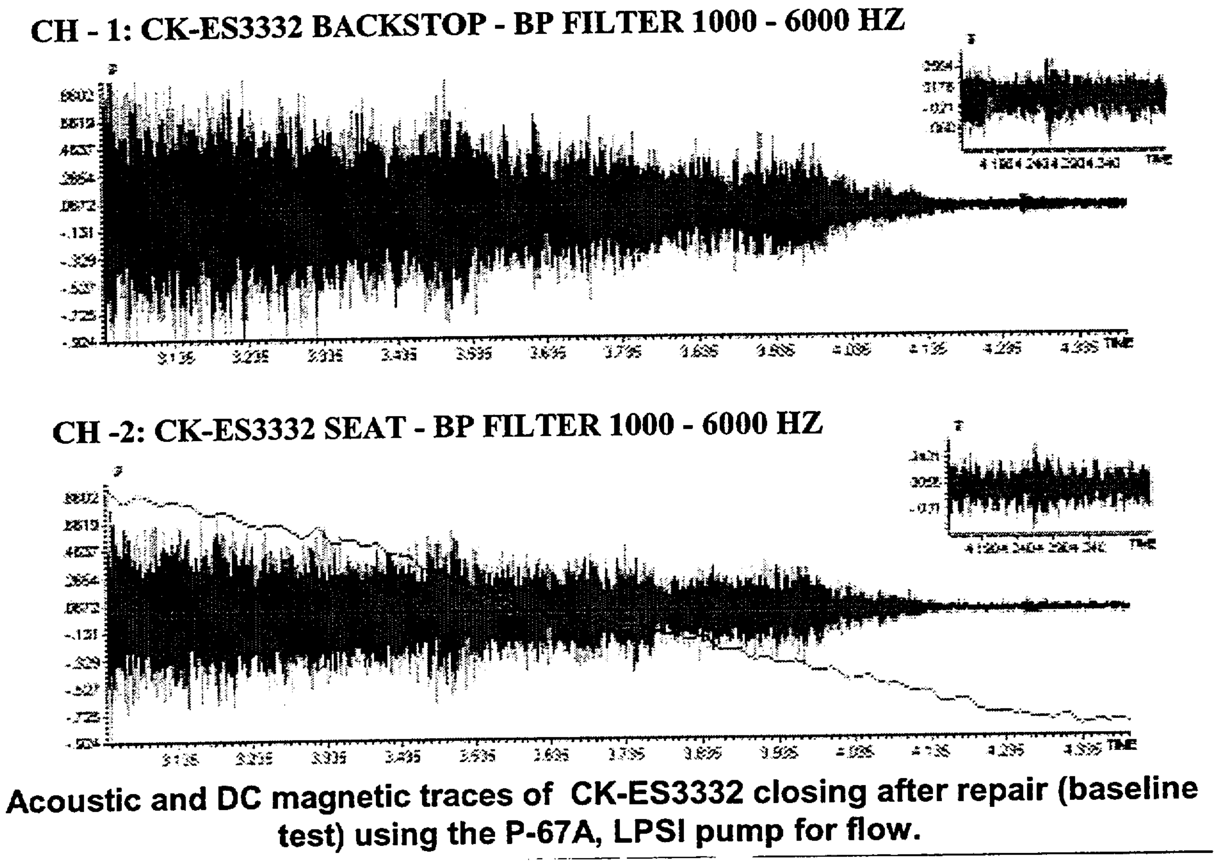
CH -2: CK-ES3332 SEAT - BP FILTER 1000 - 6000 HZ



CH - 1: CK-ES3332 FFT OF BACKSTOP - BP FILTER 1000 - 6000 HZ



Acoustic and DC magnetic traces of CK-ES3332 opening after repair (baseline test) using the P-67A, LPSI pump for flow.



Acoustic and DC magnetic traces of CK-ES3332 closing after repair (baseline test) using the P-67A, LPSI pump for flow

Nuclear Industry Check
Valve Group

June 7, 2001

Site VP

SUBJECT: The Nuclear Industry Check Valve Group (NIC) Response to: NRC INFORMATION NOTICE 2000-21

On December 15, 2000, the Nuclear Regulatory Commission (NRC) issued the subject Information Notice to Licensees. Although the Notice did not require response, the issues raised are of sufficient importance that NIC chooses to inform the members of its perspective.

NIC supports the continued use of nonintrusive testing. NIC performed, in the early 1990's, Phase 1, 2, & 3 studies that evaluated technologies that have been successfully and reliably demonstrated to assist in determining check valves operational readiness. Since then the NIC has successfully continued to demonstrate, improve, and refine the applications of these technologies.

NIC has provided various technical documents (Analysis Guide, Phase 1 through 3 reports, Flowtest, etc.) to help owners use and qualify nonintrusive technologies. These reports strongly recommend the use of multiple technologies (in combination) to provide as much information as possible about the check valves operational readiness. When multiple technologies are not possible (or results are not conclusive), then the test should be augmented with other corroborating information. This information may be in the form of indications of proper operation, past disassembly and inspection, etc.

Part of the basis for determining operational readiness is having a baseline test when the valve is known to be operating acceptably. Establishment of a baseline requires supporting information to determine the capability of the valve to perform its intended function(s).

Application of these principles when using nonintrusive testing should help improve the ability of the nuclear industry to demonstrate check valve operational readiness.

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Valve Performance Solutions Determining Frictional and Dynamic Loads from In Situ Test Evaluations

John Holstrom
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Abstract

The main focus of this presentation will be the methods that can be employed to separate frictional loads from dynamic loads in the typical industry static and dynamic testing methods and how this data can be used to predict operating loads at other dynamic conditions. A secondary focus will be on the side loads and fluidynamic lift found in large diameter, angle pattern, balanced and sleeved globe valves.

The industry generally excepted test information has been to show valve and system time history data and benchmarking specific events in the output traces such as a zero transition, seat contact, peak seating and unseating loads, torque switch trip and final output.

When these data are converted to position history rather than time history all events at all positions of travel are directly comparable. This improves the ability of the investigator to identify normal and abnormal loads, anomalies and to quantify the effects of the load and operational changes.

Resulting data can be analyzed to obtain dimensionless engineering parameters to better predict such effects as flow characteristics, side loading and fluidynamic lift (or torque).

In-Situ test data from a large diameter sleeved and balanced globe valve will be used to show how frictional loads, flow coefficients, side loading, and fluidynamic lift can be determined from observed data.

Some observed operational problems and solutions would also be provided.

Power Up-Rate Solutions MSIV Dynamic Stroke Time Evaluation

John Holstrom
Altran Corporation

Abstract

The extended power uprate of two power stations owned by Exelon Nuclear (Dresden and Quad Cities stations) involved a reanalysis of the ASME overpressure event to determine the ability to maintain the Technical Specification Safety Relief Valve Setpoint Tolerance of ±1%. This transient assumes that the reactor is operating at 102% of full power when Main Steam Isolation Valve (MSIV) closure occurs. Anticipatory scrams associated with MSIV closure is not assumed to occur. This results in a reactor scram on high reactor flux. Reactor pressure relief occurs via lifting of the safety relief valves.

When traditional analysis was applied to uprate conditions, the ±1% safety relief valve tolerance was found to be challenged. A review of the existing model found that the MSIV closure profile used on the existing transient model may have been overly conservative.

This analysis was performed to establish a realistic, yet bounding, closure profile for the MSIVs to be used in the ASME overpressure analysis. This analysis created a mathematical model of the double-acting, spring assisted actuator which included the hydraulic speed control damper to calculate a realistic relationship of valve position, and time.

The MSIV internal design was analyzed to establish the flow area at each valve position.

These products were finally combined to establish a refined flow area versus time relationship that could be used in the existing transient analysis model.

This presentation will explain the conditions that lead to the need for this approach, the methods of determining probable benefits, the basic engineering methodology, and the results from the analysis.

The model can be benchmarked against the static test stroke time data. The dynamic conditions and loads can then be added to predict stroke time changes.

Session 4

Regulatory Activities Update

Session Chair

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Air-Operated Valve Performance and Inservice Testing Issues

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Abstract

This paper discusses current regulatory activities involving the inservice testing (IST) of air-operated valves (AOVs) in nuclear power plants. The paper addresses the scope of AOVs to be included in IST programs, AOV-related Code cases approved by the NRC staff, and the status of current licensing reviews of risk-informed AOV programs.

Introduction

AOVs are used in all U.S. light-water reactor plants. They are used in a variety of applications, and the population of AOVs in each plant varies widely. The number of AOVs in a plant can be over a thousand, and the number of safety-related AOVs per plant can be several hundred. Many plants have a number of AOVs that have an important role from a risk perspective but are not designated as "safety-related." The major safety concern identified as a result of a recent NRC study (Ref. 1) from a risk perspective is the simultaneous common-cause failure of AOVs which could disable redundant trains of a system important to safety. Most of the recent NRC staff and industry attention with regard to AOVs has been focused on AOV performance.

Background

For the past several years, the NRC staff has been working with industry groups and consensus bodies to monitor the development of design basis verification and inservice testing programs for AOVs. In 1999, the NRC met with the Joint Owners' Group on Air-Operated Valves (JOG-AOV) to discuss a voluntary industry program to address AOV issues. The JOG-AOV, which was facilitated by the Nuclear Energy Institute (NEI), developed a risk-informed program (Refs. 2 & 3) that established guidance for verifying AOV performance at design-basis conditions and for performing long-term periodic verification of safety-related AOVs categorized as high-safety significant. The JOG-AOV program also provided guidance for a less-rigorous verification of AOV functionality for those AOVs determined to be low-risk significant. Although the NRC staff did not formally review nor approve the JOG-AOV program, it did provide feedback comments on the JOG-AOV program document in a letter to NEI dated October 8, 1999 (Ref. 4).

This paper was prepared by staff of the U.S. Nuclear Regulatory Commission. It may present information that does not currently represent an agreed-upon NRC staff position. NRC has neither approved nor disapproved the technical content.

On March 15, 2000, the NRC staff issued Regulatory Issue Summary (RIS) 2000-03, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," (Ref. 5). The RIS discussed the staff's intent to close out Generic Safety Issue (GSI) 158 (Ref. 6) on the basis that current regulations provide adequate requirements to ensure verification of the design-basis capability of AOVs (and other power-operated valves) and that no new regulatory requirements were needed. The RIS also noted that the NRC staff would continue to work with industry groups and to monitor licensees' activities to ensure that safety-related AOVs (and other power-operated valves) will remain capable of performing their specified functions under design-basis conditions and to provide a timely, effective, and efficient resolution of the concerns regarding AOV performance.

Inservice Testing Program Scope for Air-operated Valves (Aovs)

In establishing an IST program in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section XI (ASME Code) (Ref. 7) or the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) (Ref. 8), a question that arises frequently is, "What is the scope of AOVs that should be included in an IST program?" This question becomes more complex when a licensee is establishing a risk-informed IST program for AOVs.

The requirement for the scope of valves to be included in an IST program is addressed in Title 10 of the *Code of Federal Regulations* (10 CFR) (Ref. 9) in Section 50.55a(f). Specifically, 10 CFR 50.55a(f)(4) states, "Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the inservice test requirements...set forth in the ASME OM Code." ASME Code Class 1 valves include all valves within the reactor coolant pressure boundary. Regulatory Guide (RG) 1.26 (Ref. 10) provides guidelines for establishing the quality group classification (and ASME Code classification) for water-, steam-, and radioactive-waste-containing components in nuclear power plants other than those in the reactor coolant pressure boundary (i.e., ASME Code Class 2 and 3 components). In 10 CFR 50.55a(b)(3), the NRC incorporates by reference the ASME OM Code, 1995 Edition with the 1996 Addenda. ISTC 1.1 of the 1995 OM Code with the 1996 Addenda further defines the scope by stating that IST programs shall include active or passive valves that are required to perform a specific function in shutting down the reactor to a safe shutdown, in maintaining the safe shutdown condition, or in mitigating the consequences of an accident. The scope of the OM Code also covers pressure relief devices used for protecting systems or portions of systems that perform a required safety-related function. Therefore, the scope of valves to be included in IST programs must include ASME Code Class 1, 2, and 3 valves that are covered in ISTC 1.1 of the ASME OM Code. In addition, NUREG-1482, Section 2.2 (Ref. 11) provides guidance for selecting valves for the IST program.

Based on the above requirements and guidelines, the licensee establishes the scope of its IST program. The NRC retains the option to verify the licensees' IST program scope by inspection. Many licensees also include augmented AOVs in their IST programs. Augmented AOVs in a licensees'

IST program are AOVs which are outside the scope of the program but are included in the IST program for testing purposes. These valves are not required to meet ASME Code testing requirements.

When developing a risk-informed IST program for AOVs using Code Case OMN-12, "Alternative Requirements for Inservice Testing Using Risk Insights for Pneumatically- and Hydraulically-Operated Valve Assemblies in Light-Water Reactor Power Plants," (Ref. 12), a clear understanding of the program scope is needed for successful implementation of the program. When using this approach, the RI-IST program scope for AOVs is similar to the scope of current IST programs except that licensees must include non-ASME Code AOVs that are categorized as high-safety significant (HSS). Non-ASME Code AOVs that are categorized as low-safety significant (LSS) components are not required to be included in the RI-IST program, but if the licensee does choose to include these valve for testing purposes, they should be identified to the NRC to avoid confusion at a later date if questions arise whether they must meet ASME Code testing requirements.

NRC Draft Regulatory Guide DG-1089

On December 28, 2001, the NRC staff issued a notice in the *Federal Register* of the availability of Draft Regulatory Guide DG-1089, "Operation and Maintenance Code Case Acceptability, ASME OM Code," (Ref. 13). DG-1089 is a new proposed regulatory guide that endorses ASME OM Code Cases that have been determined by the NRC to be acceptable alternatives to the requirements of the ASME OM Code. Licensees may use the approved Code Cases without submitting a request for NRC review and approval, provided all conditions listed in the regulatory guide are followed. Use of ASME Code Cases are voluntary. However, once they are implemented, they become regulatory requirements with the same force of law as ASME OM Code requirements and NRC regulations. The draft DG-1089 was published in the *Federal Register* for public comments, and the 90 day comment period ended on March 25, 2002. The final regulatory guide will be given a new number and is scheduled to be issued in September of 2002.

Included in DG-1089 are two risk-informed Code cases of particular interest to the AOV IST programs: Code Case OMN-3, "Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of Light Water Reactor Power Plants," (Ref. 14), and Code Case OMN-12, "Alternative Requirements for Inservice Testing Using Risk Insights for Pneumatically- and Hydraulically-Operated Valve Assemblies in Light-Water Reactor Power Plants." Code Case OMN-3 establishes the methodology and process to categorize components that are part of an ASME Code risk-informed IST program into HSS and LSS components. Code Case OMN-12 establishes alternative AOV test strategies used in conjunction with Code Case OMN-3 risk-informed categorization.

Revision 0 to Code Case OMN-3 was published in the 1998 Edition of the ASME OM Code. In DG-1089, the NRC proposed four conditions on the use of Code Case OMN-3. Condition 1, which relates to program scope, specifies that HSS components must include non-ASME components categorized as HSS (this is similar to categorization of non-ASME components in Ref. 3).

Revision 0 to Code Case OMN-12 was published in 2001 Edition of the ASME OM Code. In DG-1089, the NRC proposed eight conditions on the use of Code Case OMN-12. The conditions ensure technical philosophy consistent with Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Motor-Operated Valves Assemblies in Light-Water Reactor Power Plants," (Ref. 15), developed for motor-operated valves (MOVs). The conditions proposed in DG-1089 for HSS AOVs would require licensees to (1) include a mix of static and dynamic testing that may be altered when justified by evaluation of test data, (2) evaluate within five years or three refueling outages adequacy of diagnostic test interval, (3) evaluate potential increases in core damage frequency (CDF) and risk of interval extension to ensure consistency with NRC Regulatory Guide 1.174 (Ref. 16), and (4) evaluate degradation rate and capability margin to ensure AOVs remain capable of performing their design-basis functions until the next scheduled test. The conditions proposed in DG-1089 for LSS AOVs would require that (1) AOVs remain capable of performing their design basis function until the next scheduled test, (2) setpoints are based on direct dynamic test information, a test-based methodology, or grouping with dynamically tested valves, (3) initial and periodic diagnostic tests are performed to verify setpoints, and (4) the operability of an AOV is evaluated if the valve does not satisfy the acceptance criteria.

Status of Risk Informed Air-Operated Valve Program Reviews

Two risk-informed AOV programs have been formally submitted to the NRC staff for review and approval. The licensee for the Davis-Besse nuclear power plant submitted its proposed risk-informed testing program for air-operated valves to the NRC staff in a letter dated September 11, 2000 (Ref. 17). The B&W Owners' Group Topical Report BAW-2359, "Demonstration Project to Apply Risk-Informed Inservice Testing to Air-Operated Valves," (Ref. 18), which was referenced in the Davis-Besse risk-informed AOV program was submitted to the NRC staff on July 14, 2001.

The Davis-Besse risk-informed AOV program was reviewed in detail by the staff and underwent several iterations. Due to a multitude of complications that arose in the review including higher priorities both at Davis-Besse and at the NRC, the completion of the review was delayed. In the meantime, the staff issued Draft Regulatory Guide DG-1089 as previously discussed that proposed to approve Code Case OMN-12 for RI-IST of AOVs with certain conditions. Because the final version of DG-1089 is scheduled to be issued in September 2002, rather than continue with the risk-informed AOV review, the staff and licensee mutually agreed that the most efficient and effective approach at this time was to withdraw the submittal and implement Code Case OMN-12 when DG-1089 is issued as a final regulatory guide. In this manner, the licensee may implement Code Case OMN-12 without the need for NRC staff review and approval.

The status of Topical Report BAW-2359 is uncertain at this time. The report may be overtaken by approval of Code Case OMN-12, and the need for staff review of the report may be reassessed.

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9. United States *Code of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities."

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Validation Approach for Valve Performance Prediction Methodologies

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Abstract

Since 1989, the NRC has reviewed several programs established by nuclear power plant licensees in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," GL 96-05, "Periodic Verification of the Design-Basis Capability of Safety-Related Motor-Operated Valves." During these reviews, the NRC has evaluated several methodologies developed by industry groups, individual licensees, and consultants for the prediction of the performance of valves under various system and ambient conditions. These methodologies predicted valve performance in areas such as thrust and torque requirements to open and close motor-operated valves under differential pressure and flow conditions, uncertainty in those predicted operating requirements, and the thrust required to open a valve under pressure-locking conditions. This paper provides examples of the types of methodologies for predicting valve performance that have been reviewed by the NRC, indicates the various approaches used in supporting the validation of those methodologies, and identifies key attributes to be addressed in presenting a well-supported validation of a valve performance prediction methodology.

I. Introduction

Nuclear power plant licensees, industry groups and consultants develop methodologies to provide a generic approach to address specific technical issues. The staff of the U.S. Nuclear Regulatory Commission (NRC) may review these methodologies as part of evaluations of plant-specific activities or industry-wide programs. Since 1989, the NRC has reviewed programs established by nuclear power plant licensees in response to Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," GL 96-05, "Periodic Verification of the Design-Basis Capability of Safety-Related Motor-Operated Valves." During these reviews, the NRC has evaluated several methodologies for the prediction of the performance of valves under various conditions. These methodologies predicted valve performance in areas such as thrust and torque requirements to open and close motor-operated valves (MOVs) under differential pressure and flow conditions, uncertainty in those predicted operating requirements, and the thrust required to open a valve under pressure locking conditions. In this paper, the authors discuss various methodologies for predicting valve performance that have been reviewed by the NRC. The paper includes the various approaches used in supporting the validation of those methodologies, and the key attributes to be addressed in presenting a well-supported validation of a valve performance prediction methodology.

II. Electric Power Research Institute Mov Performance Prediction Methodology

In response to weaknesses in MOV performance, the NRC issued GL 89-10 on June 28, 1989, to request that licensees ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The NRC requested that licensees complete their GL 89-10 programs within approximately three refueling outages or 5 years from the issuance of the generic letter. Subsequently, the NRC issued GL 96-05 to provide more detailed recommendations for the establishment of long-term programs to verify the design-basis capability of safety-related MOVs on a periodic basis.

In support of the effort by the nuclear industry to respond to GL 89-10, the Nuclear Energy Institute (NEI) submitted Electric Power Research Institute (EPRI) Topical Report TR-103237, "EPRI MOV Performance Prediction Program," to the NRC for its review and acceptance. EPRI developed the MOV Performance Prediction Methodology (PPM) for use by licensees in predicting the thrust and torque required to operate gate, globe, and butterfly valves under dynamic flow conditions. The EPRI MOV PPM program included the development of improved methods for prediction or evaluation of system flow parameters; gate, globe, and butterfly valve performance; and motor-actuator rate-of-loading effects (load sensitive behavior). EPRI also performed testing to evaluate parameter separately (separate effects testing) to provide information for refining the gate valve model and rate-of-loading methods; and conducted numerous MOV tests to provide data for model and method development and validation, including flow loop testing, parametric flow loop testing of butterfly valve disk designs, and plant in-situ MOV testing. EPRI integrated the individual models and methods into an overall methodology including a computer model and implementation guide.

EPRI developed the PPM from fundamental engineering principles related to MOV design and operation including consideration of fluid and friction forces. EPRI based specific aspects of the MOV PPM (such as valve internal friction coefficients) on the results of separate effects testing. EPRI validated the individual models of the MOV PPM (system, gate valve, globe valve, and butterfly valve models) using applicable data from MOV flow tests. EPRI made adjustments to the MOV PPM where determined to be appropriate based on MOV flow tests, such as including a 5% margin factor for gate valves manufactured by Borg Warner. EPRI performed an assessment of the integrated

MOV PPM using flow loop and plant in-situ test data. EPRI provided detailed documentation of the development and assessment of the methodology.

The NRC with its contractor (Idaho National Engineering and Environmental Laboratory (INEEL)) evaluated the development of the models used in the EPRI MOV PPM, the application of test data to validate those models, and the overall PPM assessment conducted by EPRI. The NRC discussed the MOV PPM with EPRI in detail and provided written questions to EPRI on the development and application of the PPM. On March 15, 1996, the NRC issued a safety evaluation (SE) finding that the EPRI MOV PPM is an acceptable methodology with certain conditions and limitations to predict the thrust or torque required to operate gate, globe, and butterfly valves within the scope of the program, and to bound the effects of load sensitive behavior on motor-actuator thrust output. On February 20, 1997, the NRC issued a supplement to the SE that accepted methods developed by EPRI for two unique gate valve designs to predict their operating thrust requirements with certain conditions and limitations.

The application of solid engineering principles with directly applicable test data represents an effective manner in which to justify a methodology. In this case, the justification for the MOV PPM by EPRI reflected a technically sound approach through the application of first engineering principles with separate effects test data used to establish reasonable values for performance parameters. By the use of first principles, EPRI was able to present a clear description of its approach and resulting methodology to licensee personnel and the NRC. The valve performance data obtained from specifically designed flow tests enabled EPRI to support the precision of its methodology in a technically defensible manner.

III. EPRI Thrust Uncertainty Method

EPRI has developed a supplemental methodology (referred to as the Thrust Uncertainty Method) in an effort to address potential conservatisms in the valve operating requirements predicted by the EPRI MOV PPM. EPRI has presented the methodology to the NRC for approval in Addendum 2 to Topical Report TR-103237-R2, "EPRI Motor-Operated Valve (MOV) Performance Prediction Program." The Thrust Uncertainty Method establishes an average conservatism in the thrust predicted by the EPRI MOV PPM to be necessary to operate gate valves under dynamic flow conditions. The Thrust Uncertainty Method then treats the conservatism as a random uncertainty that is statistically combined with other uncertainties. In this effort, EPRI compared the thrust required to operate sample gate valves during flow loop tests conducted as part of the EPRI MOV Performance Prediction Program to the thrust requirement predicted by its MOV PPM. EPRI calculated an average prediction ratio from the sample gate valves operated under either cold or hot water conditions. EPRI specifies that the Thrust Uncertainty Method is only applicable for predicting the thrust required to close gate valves.

At the outset of the review of the Thrust Uncertainty Method, the NRC noted several areas of concern regarding the acceptability of the method during a public meeting on September 20, 2000. First, if the valves used in calculating the conservatism of the EPRI MOV PPM as part of the Thrust Uncertainty Method were not fully preconditioned, the thrust required to operate those valves might increase with age. If so, the Thrust

Uncertainty Method might become inadequate to ensure the capability of those valves over time and service. Second, in that the EPRI MOV PPM was developed as a first-principles model rather than a statistical database model, it was not clear that sufficient test data are available to determine in a reliable manner the conservatism of the EPRI MOV PPM for a wide range of gate valve types and their service conditions. Third, the validation of the Thrust Uncertainty Method as described in Addendum 2 to the EPRI topical report did not provide a clear indication that the MOVs included in the validation effort would continue to be able to perform acceptably if their torque switches were set using the Thrust Uncertainty Method.

In an NEI submittal dated January 5, 2001, EPRI provided further information on its Thrust Uncertainty Method that was discussed at a public meeting on October 18, 2001. At the end of the meeting, the NRC stated that several significant concerns remain regarding the establishment and validation of the Thrust Uncertainty Method. For example, the data used in the Thrust Uncertainty Method to establish an average prediction ratio for determining a nominal value for the thrust required to close a gate valve represented a very small sample of the total population of safety-related motor-operated gate valves in the nuclear industry. Further, the non-normal distribution of the prediction ratios of the actual thrust required to close the sample gate valves under cold water conditions to the EPRI MOV PPM thrust prediction reflected a median value higher than the mean value used for the average prediction ratio in the Thrust Uncertainty Method. The NRC also noted that a significant concern existed regarding the viability of the Thrust Uncertainty Method for gate valves operated under hot water conditions because of the minimal amount of test data used in establishing an average prediction ratio.

In an NEI submittal dated December 6, 2001, EPRI indicated that several actions had been taken to help support its development and validation of the Thrust Uncertainty Method. For example, EPRI limits the Thrust Uncertainty Method to only cold water applications up to 150°F. Further, EPRI will apply the median value of the prediction ratios in predicting a nominal value for the thrust required to close a gate valve under cold water conditions as part of the Thrust Uncertainty Method. EPRI also presented additional analysis regarding the Thrust Uncertainty Method to address the remaining NRC concerns. The NRC is continuing its interaction with NEI and EPRI to complete the review of the Thrust Uncertainty Method.

IV. Pressure Locking and Thermal Binding Thrust Prediction Methodologies

On August 17, 1995, the NRC issued GL 95-07 to request that licensees perform, or confirm that they had previously performed, (1) evaluations of the operational configurations of safety-related, power-operated gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses, and any needed corrective actions, to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing the safety functions within the current licensing basis of the facility. Pressure locking can occur in flexible-wedge and double-disk gate valves when pressure in the bonnet is higher than the line pressure on both sides of a closed disk and the valve actuator is not capable of overcoming the additional thrust required as a result of the differential pressure. Thermal binding is generally associated with a solid- or flexible-wedge gate valve that is closed at high temperature and is allowed to cool before reopening is attempted such that mechanical interference occurs because of contraction of the valve body on the disk wedge.

In response to GL 95-07, many licensees used a pressure-locking methodology developed by Commonwealth Edison Company (ComEd), which is now a member of Exelon, to demonstrate that flexible wedge gate valves are capable of operating under pressure-locking conditions. In a letter to the NRC dated May 24, 1996, ComEd provided the test results from a 4-inch (1500-pound) Westinghouse valve; a 10-inch (900-pound) Crane valve; and a 10-inch (300-pound) Borg-Warner valve that were used to validate its pressure-locking methodology. A public meeting was conducted on April 9, 1997, to discuss the ComEd flexible wedge gate valve pressure locking analytical method and validation testing. In a letter to the NRC dated May 29, 1996, ComEd provided additional information on its pressure-locking methodology. After May 29, 1996, the NRC issued a number of safety evaluations on GL 95-07 submittals finding that the ComEd methodology provides a technically sound basis for assuring that valves susceptible to pressure locking are capable of performing their intended safety-related function.

The ComEd pressure-locking thrust prediction methodology is based on the Sixth Edition of *Roark's Formulas for Stress and Strain* (Young, Warren C., McGraw-Hill Book Company, New York, NY, 1989). The valve disk is assumed to act as two ideal disks connected by the hub. The differential pressure between the bonnet and the upstream side of the valve is averaged between the bonnet and the downstream side of the valve to determine a pressure locking differential pressure to be applied across the valve disks. The total stem force required to open a valve during pressure locking conditions is determined from the unwedging load, vertical pressure load, and pressure-lock load based on total contact load minus the stem rejection load.

The NRC review of the ComEd pressure locking methodology focused on the test results that were used to validate the pressure-locking methodology. The NRC verified that the quality of the testing accomplished by ComEd to validate its methodology provided meaningful and accurate test results. Actual pressure locking test results indicated that as the differential pressure between the bonnet and the downstream (or upstream) side of the valve increased, the stem thrust required to open the pressure locked valve increased. The NRC verified that the ComEd pressure-locking methodology results trended with actual pressure locking test results. The NRC also verified that actual coefficients of friction obtained during testing were used to validate the methodology. The NRC and its contractor (INEEL) tested a flexible wedge gate valve under pressure-locking conditions, and used the test results to verify that the ComEd pressure-locking methodology accurately predicted the thrust required to open the valve. The results of this testing are documented in NUREG/CR-6611, "Results of Pressure Locking and Thermal Binding Tests of Gate Valves." The NRC concluded that the ComEd pressure-locking methodology is acceptable for use provided that minimum margins are applied between calculated pressure-locking thrust and actuator capability and that diagnostic equipment accuracy and methodology limitations are applied. The NRC accepted reduced margins between calculated pressure-locking thrust and actuator capability when using an enhanced version of the ComEd methodology.

In response to GL 95-07, several licensees used a modified industry gate valve thrust equation to predict the thrust required to open flexible wedge and double disk gate valves during pressure-locking conditions. In this methodology, the total required force to operate the valve during pressure-locking conditions is the sum of the vertical forces resulting from differential-pressure loads across the two valve disks. Although a number of licensees used this methodology in their GL 95-07 submittals to the NRC, none of the licensees validated the methodology with a test program. For flexible wedge gate valves, one licensee demonstrated that results of the modified industry gate valve thrust equation were more conservative than the results obtained from the ComEd pressure locking-methodology. In its GL 95-07 submittal to the NRC, the results of the ComEd pressure locking-methodology were compared to the results of modified gate valve methodology for the same valve and pressure-locking conditions.

Pressure locking tests sponsored by the NRC were conducted by INEEL on a flexible wedge gate valve (NUREG/CR-6611). Test data demonstrated that the modified industry gate valve calculation conservatively estimated the thrust required to open a pressure-locked flexible wedge gate valve. Test data from a 4-inch Westinghouse valve and a 10- inch Crane valve were used by the NRC to demonstrate that the modified industry gate valve methodology conservatively estimated that thrust required to open a pressure-locked flexible wedge gate valve. The NRC issued a number of safety evaluations on GL 95-07 submittals finding that sizing the power actuator to satisfy the modified industry gate valve thrust equation provides a technically sound basis for assuring that flexible wedge gate valves susceptible to pressure locking are capable of performing their intended safety-related function.

Pressure-locking tests sponsored by the NRC were also conducted by INEEL on a double disk gate valve (NUREG/CR-6611). Test data demonstrated that the modified industry gate valve thrust equation underestimated the thrust required to open a pressure-locked double disk gate valve; however, the results of the equation properly trended with actual test results. The NRC issued a number of safety evaluations on GL 95-07 submittals finding that sizing the power actuator to satisfy the modified industry gate valve thrust equation provides reasonable assurance that double disk gate valves susceptible to pressure locking are capable of performing their intended safety-related function provided that there is an appropriate margin between predicted pressure-locking thrust and actuator capability. It would have been very difficult for the NRC to approve use of the modified industry gate valve thrust equation as an acceptable corrective action for pressure locking of double disk gate valves without the use of the test results in NUREG/CR-6611.

In response to GL 95-07, several licensees proposed the use of a pressure locking thrust prediction methodology that the NRC was unable to approve. The NRC review of the test data used to validate the acceptability of the proposed methodology indicated that in some instances the proposed methodology underestimated the amount of thrust required to open several different types of flexible wedge gate valves during pressure-locking conditions. Validation of the proposed pressure-locking prediction methodology became further complicated because the actual disk friction factor was not used to validate the methodology. The NRC believes that the disk friction factor is a critical parameter when validating any valve performance methodology, and it was not clear to the NRC why a generic disk friction factor was used in lieu of the actual disk friction factor to validate the proposed pressure-locking methodology. Further, actual pressure locking test results indicated that as the differential pressure between the bonnet and the downstream (or upstream) side of the valve increased, the stem thrust required to open the pressure locked valve increased. The proposed pressure-locking methodology predicted that the opposite would occur in that, as the differential pressure between the bonnet and downstream (or upstream) side of the valve increased, the stem thrust predicted to open the pressure locked valve decreased. It was not apparent to the NRC why the results of the proposed methodology were not consistent with the actual test results. Several public meetings were conducted to discuss the proposed pressure-locking thrust prediction methodology, and additional information on the proposed pressure locking method was provided in several letters to the NRC. As a result, the NRC was unable to approve the proposed pressure-locking methodology, and licensees used other methods to demonstrate that valves were capable opening during pressure-locking conditions.

In response to GL 95-07, other licensees proposed the use of a thermal binding or pressure-locking thrust prediction methodologies that were developed to calculate the thrust required to open valves during thermal-binding or pressure-locking conditions. However, adequate test data were not available to the NRC to evaluate the licensee's thrust prediction methodologies. Methods other than the proposed thermal-binding or pressure locking methodology were used to demonstrate that valves were capable of opening during thermal-binding or pressure-locking conditions.

V. Conclusion

The application of solid engineering principles with directly applicable test data represents an effective manner in which to justify a methodology. Actual test valve parameters such as disk friction factor, packing load, stem thrust, test pressures and valve characteristics should be used in the validation process whenever possible. Any inconsistencies or anomalies between actual test results and the methodology should be understood and thoroughly explained. Typically, it is not feasible for the NRC to review methodologies as part of plant inspection activities because methodologies are generally too complex to perform a sufficiently detailed review during the time period allotted for inspection activities unless prior arrangements are made. Licensees should work with their owners groups or NRC project manager to determine the most efficient approach in obtaining NRC acceptance of methodologies developed to address specific technical issues.

Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves

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Abstract

Many fluid systems at nuclear power plants depend on the successful operation of motor-operated valves (MOVs) in performing system safety functions. As a result of problems identified in the 1980s with MOV performance at nuclear power plants, the NRC issued Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of the Design-Basis Capability of Safety-Related Motor-Operated Valves," requesting that nuclear power plant licensees verify initially and periodically the design-basis capability of MOVs in safety-related systems. In response to GL 96-05, the nuclear power plant owners groups developed an industry-wide Joint Owners Group (JOG) program for periodic verification of the design-basis capability of safety-related MOVs. In a safety evaluation, the NRC accepted the JOG program as an industry-wide response to GL 96-05 with respect to age-related valve degradation. The NRC issued GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," requesting that licensees ensure that safety-related power-operated gate valves susceptible to pressure locking or thermal binding are capable of performing their safety functions. Licensees of all active operating reactor units have completed their programs to verify initially the design-basis capability of safety-related MOVs in response to GL 89-10, and to address potential pressure locking and thermal binding of safety-related power-operated valves in response to GL 95-07. Licensees are currently implementing their long-term MOV programs in response to GL 96-05. The NRC staff has completed its review of GL 96-05 programs established at individual nuclear plants through significant reliance on licensee commitments to implement the JOG program on MOV periodic verification. This paper discusses NRC staff activities regarding the periodic verification of the design-basis capability of safety-related MOVs, and monitoring of the nuclear industry's activities to ensure proper performance of safety-related MOVs.

I. Introduction

Many fluid systems at nuclear power plants depend on the successful operation of motor-operated valves (MOVs) in performing their system safety functions. MOVs must be capable of operating under design-basis conditions, which may include high differential pressure and flow, high ambient temperature, and degraded motor voltage. The design of the MOV must apply valid engineering equations and parameters to ensure that the MOV will operate as intended during normal plant operations and design-basis events. Manufacturing, installation, preoperational testing, operation, inservice testing (IST), maintenance, and replacement must be conducted by trained personnel using proper procedures. Surveillance must be performed and testing criteria must be applied on a soundly based frequency in a manner that suitably detects questionable operability or degradation. Moreover, these activities must be monitored by a strong quality assurance program.

The regulations of the U.S. Nuclear Regulatory Commission (NRC) require that components that are important to the safe operation of a U.S. nuclear power plant be treated in a manner that ensures their performance. Appendix A, "General Design Criteria for Nuclear Power Plants," and Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) contain broadly based requirements in this regard. In 10 CFR 50.55a, the NRC has required U.S. nuclear power plant licensees to implement provisions of the American Society of Mechanical Engineers (ASME) *Boiler & Pressure Vessel Code* (B&PV Code) for testing of MOVs as part of their IST programs. On September 22, 1999, the NRC revised 10 CFR 50.55a to require licensees implementing the 1995 Edition with the 1996 Addenda of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) to supplement the quarterly MOV stroke-time testing specified in the ASME Code with a program to verify MOV design-basis capability on a periodic basis.

Operating experience at nuclear power plants in the 1980s and 1990s revealed weaknesses in many activities associated with MOV performance. For example, some engineering analyses used in the original sizing and setting of MOVs did not adequately predict the thrust and torque required to open and close valves under design-basis conditions. Both regulatory and industry research programs later confirmed the weakness in the initial design and qualification of MOVs. For example, the NRC Office of Nuclear Regulatory Research sponsored an extensive program at the Idaho National Engineering and Environmental Laboratory (INEEL) to study the performance of MOVs under various flow, temperature, and voltage conditions. In addition, the nuclear industry sponsored a significant program by the Electric Power Research Institute (EPRI) to develop a computer methodology to predict the performance of MOVs under a wide range of operating conditions. Poor MOV performance also resulted from shortcomings in maintenance programs, such as inadequate procedures and training. Further, testing of MOVs to measure valve stroke times under zero differential-pressure and flow conditions was shown not to detect certain deficiencies that could prevent MOVs from performing their safety functions under design-basis conditions.

II. Verification of MOV Design-Basis Capability

In response to weaknesses in MOV performance, the NRC staff issued Generic Letter (GL) 89-10 (June 28, 1989), "Safety-Related Motor-Operated Valve Testing and

Surveillance." In GL 89-10, the NRC staff requested that licensees ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically, testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. The NRC staff requested that licensees complete their GL 89-10 programs within approximately three refueling outages or 5 years of the issuance of the generic letter.

In support of the regulatory activities to ensure MOV design-basis capability, the NRC Office of Nuclear Regulatory Research identified areas in which research and analysis were required to assist in evaluating MOV programs at nuclear power plants. For example, the NRC performed research to evaluate (1) performance of MOVs under pump flow and blowdown conditions; (2) output of ac-powered and dc-powered MOV motor actuators; (3) the increase in friction of aged samples of valve materials; (4) methods to determine appropriate values for stem friction coefficient; (5) pressure locking and thermal binding of gate valves; and (6) the effect of ambient temperature on stem lubricant performance. For example, the NRC sponsored flow testing of several MOVs by INEEL under normal flow and blowdown conditions. The testing revealed that (1) more thrust was required to operate gate valves than predicted by standard industry methods; (2) some valves were internally damaged under blowdown conditions and their operating requirements were unpredictable; (3) static and low flow testing might not predict valve performance under design-basis flow conditions; (4) during valve opening strokes, the highest thrust requirements might occur at unseating or in the flow stream; (5) partial valve stroking did not reveal the total thrust required to operate the valve; (6) torque, thrust, and motor operating parameters were needed to fully characterize MOV performance; and (7) reliable use of MOV diagnostic data requires accurate equipment and trained personnel. The NRC provided detailed test results in NUREG/CR-5406 (October 1989), "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test;" NUREG/CR-5558 (January 1991), "Generic Issue 87: Flexible Wedge Gate Valve Test Program;" NUREG/CR-5720 (June 1992), "Motor-Operated Valve Research Update;" and NUREG/CR-6100 (September 1995), "Gate Valve and Motor-Operator Research Findings." The NRC summarizes some of the results of the MOV research program in NRC Information Notice 90-40 (June 5, 1990), "Results of NRC-Sponsored Testing of Motor-Operated Valves." Additional examples of MOV research sponsored by the NRC are discussed later in this paper.

To assist nuclear power plant licensees in responding to GL 89-10, EPRI developed the MOV Performance Prediction Methodology (PPM) to determine dynamic thrust and torque requirements for gate, globe, and butterfly valves based on first-principles of MOV design and operation. EPRI described the methodology in Topical Report TR-103237 (Revision 2, April 1997), "EPRI MOV Performance Prediction Program." The EPRI MOV PPM program included the development of improved methods for prediction and evaluation of system flow parameters; gate, globe, and butterfly valve performance; and motor-actuator rate-of-loading effects (load sensitive behavior). EPRI also performed separate effects testing to provide information for refining the gate valve model and rate-of-loading methods; and conducted numerous MOV tests to provide data for development and validation of the models and methods, including flow loop testing, parametric flow loop testing of butterfly valve disk designs, and in-situ MOV testing. EPRI integrated the individual models and methods into an overall methodology including a computer model and implementation guide. On March 15, 1996, the NRC staff issued a safety evaluation (SE) accepting the EPRI MOV PPM with certain conditions and limitations. On February 20, 1997, the staff issued a supplement to the SE on general issues and two unique gate valve designs. On April 20, 2001, the staff issued Supplement 2 to the SE addressing an update of the computer model.

NRC Information Notice (IN) 96-48 (August 21, 1996), "Motor-Operated Valve Performance Issues," alerted licensees to lessons learned from the EPRI MOV program. Among the lessons learned were: (1) the thrust requirements to operate some gate valves under pump flow and blowdown conditions were higher than predicted by the valve manufacturers; (2) a potential exists for gate valves to be damaged when operating under blowdown conditions such that the thrust requirements can be unpredictable; (3) the effective flow area in some globe valves can be larger than expected and can cause thrust requirements to be higher than predicted; and (4) the friction coefficients for sliding surfaces in gate valves can increase with service before reaching a plateau. In IN 96-48, the staff noted that some of the EPRI information is applicable to gate, globe, and butterfly valves regardless of the type of actuator operating the valve.

Nuclear power plant licensees implemented the recommendations of GL 89-10 through a combination of design-basis reviews, revision of MOV calculations and procedures, static and dynamic diagnostic testing, industry-sponsored research programs, and trending of test results. The industry expended significant resources to resolve the deficiencies in the design, qualification, and application of safety-related MOVs that led to the issuance of GL 89-10. The results of the GL 89-10 programs and their implementation include (1) MOV sizing calculations and switch settings have been revised to reflect actual valve performance; (2) improved valve performance prediction methods have been developed; (3) valve internal dimensions are being addressed to provide assurance of predictable gate valve performance under blowdown conditions; (4) friction coefficients in new or refurbished gate valves have been found to increase with service until a plateau reached; (5) MOV output prediction methods have been updated; and (6) personnel training and maintenance practices have been improved. The NRC staff has evaluated the MOV program at each nuclear plant through onsite inspections of the design-basis capability of safety-related MOVs. The NRC staff has closed its review of GL 89-10 for each active U.S. nuclear power plant.

III. Long-term Aspects of MOV Performance

On September 18, 1996, the NRC staff issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," to provide recommendations for assuring the capability of safety-related MOVs to perform their design-basis functions over the long term. In GL 96-05, the NRC staff requested that licensees establish a program, or ensure the effectiveness of their current program, to verify on a periodic basis that safety-related MOVs continue to be capable of performing their safety functions within the current licensing basis of the facility. The guidance in GL 96-05 supersedes the guidance in GL 89-10 on long-term MOV programs.

In GL 96-05, the NRC staff noted five attributes of effective programs for periodic verification of safety-related MOV design-basis capability at nuclear power plants:

(1) A risk-informed approach may be used to prioritize valve test activities, such as frequency of individual valve tests and selection of valves to be tested.

(2) The valve test program provides adequate confidence that safety-related MOVs will remain operable until the next scheduled test.

(3) The importance of the valve is considered in determining an appropriate mix of exercising and diagnostic testing. In establishing the mix of testing, the benefits (such as identification of decreased thrust output and increased thrust requirements) and potential adverse effects (such as accelerated aging or valve damage) are considered when determining the appropriate type of periodic verification testing for each safety-related MOV.

(4) All safety-related MOVs covered by the GL 89-10 program are considered in the development of the periodic verification program. The program includes safety-related MOVs that are assumed to be capable of returning to their safety position when placed in a position that prevents their safety system (or train) from performing its safety function; and the system (or train) is not declared inoperable when the MOVs are in their nonsafety position.

(5) Valve performance and maintenance are evaluated and monitored, and the periodic verification program is periodically adjusted as appropriate.

In response to GL 96-05, nuclear power plant owners groups developed an industry-wide Joint Owners Group (JOG) Program on MOV Periodic Verification to obtain benefits from sharing information between licensees on MOV performance. The participating owners groups are the Boiling Water Reactor Owners Group (BWROG), the Babcock & Wilcox Owners Group (B&WOG), the Combustion Engineering Owners Group (CEOG), and the Westinghouse Owners Group (WOG). Elements of the JOG program include (1) an "interim" MOV periodic verification program for applicable licensees to use in response to GL 96-05; (2) a 5-year dynamic testing program to identify potential age-related increases in required thrust and torque to operate gate, globe, and butterfly valves under dynamic conditions; and (3) a long-term MOV diagnostic program to be based on information from the dynamic testing program. On October 30, 1997, the NRC staff issued an SE accepting the JOG Program on MOV Periodic Verification with certain conditions and limitations. Most licensees committed to implement the JOG program as part of their response to GL 96-05.

The NRC staff meets periodically with JOG to discuss the status and results of the JOG program. General observations to date from the JOG program include (1) the dominant influence for valve factor increase in gate valves is disassembly and reassembly of valves prior to testing; (2) for non-disassembled gate valves, initially low valve factors tend to increase and high valve factors remain stable or decrease; (3) bearing friction degradation was not identified for butterfly valves with bronze bearings in treated water, or with non-bronze bearings in treated or untreated water systems; (4) significant variation was found in bearing friction for butterfly valves with bronze bearings in untreated water systems; (5) balanced disk globe valves demonstrated stable valve factors; and (6) unbalanced disk globe valves demonstrated only small changes in valve factor. The JOG dynamic test program is scheduled to be completed in October 2002, but a few dynamic tests will be conducted after that date. JOG plans to submit a revised topical report describing the long-term MOV periodic verification program following its evaluation of the MOV dynamic test program results. The NRC staff intends to prepare a supplement to the SE on the JOG program upon review of the revised topical report.

Licensees are applying risk insights in implementing their long-term MOV programs. In Topical Report NEDC 32264, "Application of Probabilistic Safety Assessment to Generic Letter 89-10 Implementation," BWROG describes a methodology to rank MOVs according to their relative importance to core damage frequency and other considerations to be applied by an expert panel. On February 27, 1996, the NRC staff issued an SE accepting the BWROG methodology for risk ranking MOVs with certain conditions and limitations. On June 2, 1997, WOG submitted Engineering Report V-EC-1658 (Revision 1) describing an MOV risk-ranking approach for Westinghouse-design nuclear plants. On April 14, 1998, the NRC staff issued an SE accepting the WOG methodology for risk ranking MOVs with certain conditions and limitations.

As the JOG program focuses on potential increases in MOV operating requirements, licensees address potential degradation in the output of MOV motor actuators by their plant-specific programs. In the late 1990s, the NRC sponsored research at INEEL to study the performance of ac-powered MOV motor actuators manufactured by Limitorque Corporation, under various temperature and voltage conditions. For the Limitorque ac-powered motor-actuator combinations tested, the research indicated that (1) actuator efficiency might not be maintained at "run" efficiency published by the manufacturer; (2) degraded voltage effects can be greater than predicted by the square of the ratio of actual to rated motor voltage; (3) some motors produce more torque output than predicted by their nameplate rating; and (4) temperature effects on motor performance appeared consistent with the Limitorque guidance. The NRC study of ac-powered MOV output is described in NUREG/CR-6478 (July 1997), "Motor-Operated Valve (MOV) Actuator Motor and Gearbox Testing." The nuclear industry also evaluated the output capability of ac-powered MOVs at several plants. In response to the new information on ac-powered MOV performance, Limitorque provided updated guidance in its Technical Update 98-01 (May 15, 1998) and Supplement 1 (July 17, 1998) for the prediction of ac-powered MOV motor actuator. The NRC alerted licensees to the new information on ac-powered MOV output in Supplement 1 (July 24, 1998) to Information Notice 96-48. In its technical update, Limitorque also indicated that updated guidance for predicting the output capability of dc-powered motor actuators would be issued.

Following the NRC review of ac-powered MOV performance, the NRC sponsored research at INEEL to study the performance of Limitorque dc-powered MOV motor actuators under various temperature and voltage conditions. For the Limitorque dc-powered motor-actuator combinations tested, the research indicated that (1) ambient temperature effects were more significant than predicted; (2) use of a linear voltage factor needs to consider reduced speed, increased motor temperature, and reduced motor output; (3) stroke-time increase is significant for some dc-powered MOVs under loaded conditions; and (4) actuator efficiency may fall below the published "pullout" efficiency at low speed and high load conditions. The research results are provided in NUREG/CR-6620 (May 1999), "Testing of dc-Powered Actuators for Motor-Operated Valves."

On June 23, 2000, the BWROG forwarded Topical Report NEDC-32958 (March 2000), "BWR Owners' Group dc Motor Performance Methodology - Predicting Capability and Stroke Time in dc Motor-Operated Valves," to the NRC staff for information. On October 2, 2000, the BWROG recommended an implementation schedule of 12 months or the first refueling outage (whichever is later) for first priority MOVs (those with one- or two-cycle JOG static test frequencies), and two refueling outages for second priority MOVs (remaining GL 96-05 MOVs) with a start date of when the NRC acknowledged the methodology. On August 1, 2001, the NRC issued Regulatory Issue Summary (RIS) 2001-15, "Performance of dc-powered Motor-Operated Valve Actuators," that informs licensees of the availability of improved industry guidance for predicting dc-powered MOV actuator performance. In RIS 2001-15, the NRC staff stated that, based on a sample review, the BWROG methodology represents a reasonable approach to improvement of past industry guidance for predicting dc-powered MOV stroke time and output. The staff considers the BWROG methodology to be applicable to Boiling Water Reactor (BWR) and Pressurized Water Reactor plants because of similarity in the design and application of dc-powered MOVs. With the availability of the new BWROG methodology, the staff considers that the regulatory issue of adequate prediction of dc-powered MOV performance can be effectively resolved through implementation of improved industry guidance.

In support of the NRC review of the JOG program, the NRC has sponsored studies at INEEL and Battelle Institute in Columbus, Ohio, of the effects of aging on Stellite 6 which is used on sliding friction surfaces in valves. The tests of specimens in environments of temperature, pressure, and water chemistry typical of BWR nuclear plants were intended to determine the effects of film buildup on seating surfaces and the impact of the film on valve performance. The test results indicated that friction coefficients continue to increase with film thickness and that friction coefficients decrease with subsequent valve strokes. For one selected test, specimens subjected to prior periodic strokes demonstrated a lower trend in the friction coefficients than those specimens that were not subject to periodic strokes. An independent evaluation of test results indicated that the trends were valid, but that more data are needed to obtain precise conclusions. The test results are provided in INEEL/EXT-99-00116 (April 1999), "Summary and Evaluation of NRC-Sponsored Stellite 6 Aging and Friction Tests." The NRC is conducting limited additional research to verify the overall program results.

To provide additional support for the NRC review of long-term MOV programs, the NRC is sponsoring an ongoing study at INEEL of the aging of stem lubricants and the effects of ambient temperature on their lubricating properties. Results to date have indicated that the stem friction coefficient for some lubricants can increase significantly under high ambient temperature conditions. The resulting increased stem friction coefficient can cause a loss in the thrust delivered by the MOV motor actuator. The NRC summarizes the current results of the research in NUREG/CR-6750 (October 2001), "Performance of MOV Stem Lubricants at Elevated Temperature."

Each U.S. nuclear power plant licensee submitted a description of plans for periodic verification of the design-basis capability of safety-related MOVs in response to GL 96-05. The NRC staff reviewed the licensee submittals and conducted inspections of GL 96-05 programs at a sample of nuclear plants. The staff prepared an SE to document its review of the response to GL 96-05 by each licensee. Where a licensee committed to implement the JOG program, the NRC staff relied to a significant extent on that commitment in preparing the SE without the need for plant-specific inspection activity in most instances. The NRC staff reviewed GL 96-05 programs of licensees that did not commit to the JOG program by a separate process of submittals and inspections, as appropriate. The NRC has completed its review of GL 96-05 programs for each active U.S. nuclear power plant. The NRC will monitor the long-term MOV programs at U.S. nuclear plants using Inspection Procedure 62708, "Motor-Operated Valve Capability," as part of the NRC reactor oversight program.

IV. ASME Code Improvements for MOV Inservice Testing

The ASME Code specifies that stroke-time testing of MOVs be conducted as part of the IST programs of nuclear power plants on a quarterly frequency where practical. The NRC and the industry have long recognized the limitations of stroke-time testing as a means of assessing the operational readiness of MOVs to perform their design-basis safety functions. In the most recent revision to 10 CFR 50.55a, the NRC requires U.S. nuclear power plant licensees implementing the 1995 Edition with the 1996 Addenda of the ASME OM Code to supplement the quarterly MOV stroke-time testing specified in the Code with a program to verify MOV design-basis capability on a periodic basis. In the *Federal Register* notice (64 FR 51370) issuing the rule, the NRC discusses the implementation of MOV programs in response to GL 89-10 and GL 96-05 at nuclear power plants, and the requirement to supplement MOV stroke-time testing.

In response to concerns regarding the adequacy of MOV stroke-time testing, the ASME Operations and Maintenance Code Committee developed performance-based ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants, OM Code 1995 Edition; Subsection ISTC." As an alternative to quarterly stroke-time testing, ASME Code Case OMN-1 allows periodic exercising of all safety-related MOVs once per refueling cycle and periodic diagnostic testing under static or dynamic conditions, as appropriate, on a frequency determined by MOV performance in terms of margin and degradation rate. In GL 96-05, the NRC staff noted that the method in ASME Code Case OMN-1 could be used as part of a licensee's response to the generic letter.

In the regulations, the NRC endorsed the use of ASME Code Case OMN-1 as an acceptable alternative to the quarterly MOV stroke-time testing specified in the ASME OM Code with certain conditions. The NRC stated that, where a selected test interval for an MOV under ASME Code Case OMN-1 exceeds 5 years, the licensee must evaluate information obtained from valve testing during the initial 5-year period to validate assumptions made in justifying the longer test interval. The NRC also specified that licensees must evaluate the potential increase in risk associated with extending the quarterly exercise frequency for MOVs identified as having a high safety significance. In the *Federal Register* notice, the NRC indicated that, as part of implementing ASME Code Case OMN-1, licensees need to consider the benefits (such as identification of decreased thrust output and increased thrust requirements) and potential adverse effects (such as accelerated aging or valve damage) when determining appropriate testing for each MOV. Also, the NRC noted that the provisions of ASME Code Case OMN-1 would satisfy the regulatory requirements for supplementing quarterly MOV stroke-time testing with the conditions specified in the rule.

The NRC staff has granted requests from several nuclear power plant licensees to apply performance-based ASME Code Case OMN-1 as an alternative to the quarterly MOV stroke-time testing in their particular ASME Code of record. The NRC staff is completing a regulatory guide that proposes to accept on a generic basis the use of ASME Code Case OMN-1 as an alternative to the MOV stroke-time test provisions of the ASME Code with certain conditions. The regulatory guide also proposes to accept ASME Code Case OMN-11, "Risk-Informed Testing of Motor-Operated Valves," with certain conditions that, when implemented in conjunction with Code Case OMN-1, provides emphasis on high-risk MOVs with relaxation of the test provisions for low-risk MOVs. Over the longer term, ASME is preparing a mandatory appendix to replace the quarterly MOV stroke-time testing specified in the ASME Code with performance-based provisions similar to those in ASME Code Case OMN-1.

V. Pressure Locking and Thermal Binding of Gate Valves

One typical method that "pressure locking" can occur in flexible-wedge and double-disk gate valves is when pressure in the bonnet is higher than the line pressure on both sides of a closed disk and the valve actuator is not capable of overcoming the additional thrust required as a result of the differential pressure. Thermal binding is generally associated with a solid- or flexible-wedge gate valve that is closed at high temperature and is allowed to cool before reopening is attempted such that mechanical interference occurs because of contraction of the valve body on the disk wedge. On August 17, 1995, the NRC issued GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," to request that licensees perform, or confirm that they had previously performed, (1) evaluations of the operational configurations of safety-related, power-operated (including motor-, air-, and hydraulically operated) gate valves for susceptibility to pressure locking and thermal binding; and (2) further analyses, and any needed corrective actions, to ensure that safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding are capable of performing their safety functions within the current licensing basis of the facility.

NUREG/CR-6611 (May 1998), "Results of Pressure Locking and Thermal Binding Tests of Gate Valves," describes testing sponsored by the NRC Office of Nuclear Regulatory Research at INEEL to study pressure locking and thermal binding of gate valves. The test valves included a six-inch Walworth flexible-wedge gate valve and a six-inch Anchor/Darling double-disc gate valve. Both valves were determined to be susceptible to pressure locking. During the INEEL testing, heatup of the valve caused the bonnet to pressurize slowly until leakage was overcome and then to pressurize rapidly. Air pockets were found to remain trapped in the valve bonnet after both heatup and subsequent cooldown. No significant increase in thrust requirements was found during thermal binding tests for these valves. A previous test program had revealed a significant increase in unseating load under thermal binding conditions.

In reviewing the response of each licensee to GL 95-07, the NRC staff determined whether the licensee had performed appropriate evaluations of the operational configurations of safety-related power-operated gate valves to identify valves that are susceptible to pressure locking or thermal binding. The staff then determined whether the licensee had taken, or was scheduled to take, the appropriate corrective actions to ensure that these valves are capable of performing their intended safety functions. As part of its review, the staff evaluated methodologies developed by licensees to predict the thrust required to open flexible-wedge gate valves under pressure locking conditions. The NRC staff has completed its review of licensee responses to GL 95-07 through issuance of an SE addressing each active U.S. nuclear power plant.

VI. Conclusions

As a result of problems identified in the 1980s with MOV performance at nuclear power plants, the NRC issued GLs 89-10 and 96-05 requesting that licensees verify initially and periodically the design-basis capability of MOVs in safety-related systems at nuclear power plants. In response to GL 96-05, the nuclear power plant owners groups developed an industry-wide JOG program for periodic verification of the design-basis capability of safety-related MOVs. The NRC accepted the JOG program as an industry-wide response to GL 96-05 with respect to age-related valve degradation. The NRC issued GL 95-07 requesting that licensees ensure that safety-related power-operated gate valves susceptible to pressure locking or thermal binding are capable of performing their safety functions. Licensees of all active operating reactor units have completed their programs to verify initially the design-basis capability of safety-related MOVs in response to GL 89-10, and to address potential pressure locking and thermal binding of safety-related power-operated valves in response to GL 95-07. Licensees are currently implementing their long-term MOV programs in response to GL 96-05. The NRC staff has completed its review of GL 96-05 programs established at individual nuclear plants through significant reliance on licensee commitments to implement the JOG program on MOV periodic verification. In its regulations, the NRC has directed licensees implementing the ASME OM Code to supplement the quarterly MOV stroke-time testing in their IST programs with a program to periodically verify MOV design-basis capability. The NRC staff has granted requests from several licensees to apply performance-based ASME Code Case OMN-1 as an alternative to the quarterly MOV stroke-time testing in their ASME Code of record. In its regulations, the NRC has accepted the use of ASME Code OMN-1 as an alternative to MOV stroke-time testing for licensees implementing the ASME OM Code. The NRC staff is preparing a regulatory guide that proposes to accept on a generic basis ASME Code Cases OMN-1 and OMN-11 for performance-based approaches to MOV testing together with the application of risk insights. The NRC continues to monitor licensee activities related to the performance of safety-related MOVs through the reactor oversight program.

VII. References

ASME, *Boiler and Pressure Vessel Code.*

ASME, *Code for Operation and Maintenance of Nuclear Power Plants.*

ASME, Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants, OM Code 1995 Edition; Subsection ISTC."

ASME, Code Case OMN-11, "Risk-Informed Testing of Motor-Operated Valves."

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Rulemaking Activities on Inservice Testing

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Abstract

The U.S. Nuclear Regulatory Commission (NRC) regulations in Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a) establishes requirements for the application of codes and standards in the performance of inservice inspection and testing of components used in U.S. nuclear power plants. The NRC periodically updates 10 CFR 50.55a to incorporate by reference recent editions and addenda to the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) for inservice testing of pumps and valves used in U.S. nuclear power plants. The NRC is currently updating 10 CFR 50.55a to incorporate by reference a recent edition to the ASME OM Code. Further, the NRC is revising the previous approach in referencing ASME Code Cases for use by nuclear power plant licensees as acceptable alternatives to the provisions of the ASME OM Code. This paper will present the status of current rulemakings and future rulemaking plans related to inservice testing of pumps and valves; key aspects of recent rulemakings to incorporate by reference the ASME Code; the revised NRC approach for referencing ASME Code Cases; and NRC endorsement of significant new Code Cases.

I. Incorporation By Reference A Later Edition and Addenda of ASME Code

On August 3, 2001(66 FR 40626), the NRC published a proposed rule in the *Federal Register* that presented an amendment to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," that would have revised the requirements for construction, inservice inspection (ISI), and inservice testing (IST) of nuclear power plant components. For construction, the proposed amendment would have permitted the use of the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III, Division 1, of the ASME Boiler and Pressure Vessel (BPV) Code for Class 1, Class 2, and Class 3 components with no new modifications or limitations. For ISI, the proposed amendment would have required licensees to implement the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV

This paper was prepared by staff of the U.S. Nuclear Regulatory Commission. It may present information that does not currently represent an agreed-upon NRC staff position. NRC has neither approved nor disapproved the technical content.

Code, for Class 1, Class 2, Class 3, Class MC, and Class CC components with modifications and limitations. For IST, the proposed amendment would have required licensees to implement the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code for Class 1, Class 2, and Class 3 pumps and valves with one new modification.

Interested parties were invited to submit written comments for consideration on the proposed rule. Comments were received from 17 separate sources on the proposed rule. These sources consisted of 10 utilities, 4 service organizations, and 3 individuals. In consideration of the public comments, the NRC deleted or revised a number of modifications and limitations that were in the proposed rule in this final rule. The following public comments on the proposed rule pertain to the ASME OM Code.

Comments on OM Code

Although the technical requirements in 10 CFR 50.55a(b)(3)(ii) were not revised in the proposed rule, several commenters stated that the reference to motor-operated valve (MOV) stroke-time testing in the existing 10 CFR 50.55a(b)(3)(ii) is confusing because there are other MOV test requirements in the ASME OM Code (such as position indication and seat leakage testing) that are applicable in addition to stroke-time testing. The commenters suggested that a licensee might incorrectly interpret 50.55a(b)(3)(ii) as requiring that only MOV stroke-time testing be performed in accordance with the OM Code. The NRC believes the current regulation in 10 CFR 50.55a(b)(3)(ii) clearly states that licensees must meet all of the ASME Code provisions for testing MOVs. The NRC is not aware of any misunderstanding among licensees regarding the intent of the regulatory requirement for MOVs. However, to avoid any potential confusion in the future, 10 CFR 50.55a(b)(3)(ii) is being revised to clarify that licensees must comply with the provisions of the ASME OM ISTC Code for testing MOVs.

10 CFR 50.55a(b)(3)(vi) in the proposed rule would have required an exercise interval of 2 years for manual valves within the scope of the ASME OM Code in lieu of the exercise interval of 5 years specified in the 1999 Addenda and the 2000 Addenda of the ASME OM Code. The 1998 Edition of the ASME OM Code specified an exercise interval of 3 months for manual valves within the scope of the Code. The 1999 Addenda to the ASME OM Code revised ISTC-3540 to extend the exercise frequency for manual valves to 5 years, provided that adverse conditions do not require more frequent testing. A number of commenters stated that 10 CFR 50.55a(b)(3)(vi) in the proposed rule should be withdrawn because sufficient justification exists to allow the extension of the exercise interval for manual valves to 5 years. The justification for the 5-year frequency is the simplicity of manual valves (limited number of failure causes) and that the ASME OM Code allows other valves (safety and relief valves) to be tested on a 5-year or longer frequencies. The NRC believes there is a lack of operational data or experience to allow extending the exercise interval for manual valves to 5 years. The NRC review of licensee IST programs indicates that manual valves are exercised every 3 months except in instances where it is impractical to operate valves during unit operation. Valves are then exercised when the unit is in a cold shutdown condition, and the exercise frequency cannot exceed 2 years. Therefore, a 2-year interval for exercising manual valves is justified because the available manual valve exercise data supports the 2-year interval. The NRC

has approved longer test intervals for other types of valves in the ASME OM Code but the longer test intervals include additional means to determine component degradation. For example, although the ASME OM Code test strategy for Class 2 and 3 relief valves has a testing interval of 10 years, Class 2 and 3 relief valves are subject to grouping and sample expansion if there is a test failure. Manual valves that are required to be exercised are not subject grouping and sample expansion. Furthermore, obstruction from silting or blockage, or corrosion of valve internals are possible failure modes for safety-related manual valves that are not applicable to other types of valves with longer test intervals. Exercising manual valves minimizes both of these failure modes and also allows for more immediate detection if an obstruction or corrosion induced failure occurs.

Comments on Use of Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Public Law (Pub. L.) 104-113, requires agencies to use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or is otherwise impractical. A number of commenters stated that the NRC approval of the ASME Code with exceptions (i.e., modifications and limitations) does not meet the spirit of Pub. L. 104-113. Although Pub. L. 104-113 requires Federal agencies to use industry consensus standards to the extent practical, it does not require Federal agencies to endorse a standard in its entirety, nor does it forbid Federal agencies from endorsing industry consensus standards with limitations or modifications. The law does not prohibit an agency from generally adopting a voluntary consensus standard while taking exception to specific portions of the standard if those provisions are deemed to be "inconsistent with applicable law or otherwise impractical." Furthermore, taking specific exceptions furthers the Congressional intent of Federal reliance on voluntary consensus standards because it allows the adoption of substantial portions of consensus standards without the need to reject the standards in their entirety because of limited provisions which are not acceptable to the agency. Moreover, there is no legislative history suggesting that Congress intended agencies to take an "all or nothing" approach to endorsement of voluntary consensus standards under the Act, and the OMB guidance implementing Pub. L. 104-113 does not address the matter. Finally, there is legislative history on Pub. L. 104-113 indicating that Congress did not intend each agency to prepare lengthy reports justifying the agency's decision not to adopt a voluntary consensus standard, much less an in-depth report detailing the reasons for each modification or limitation that an agency imposes on the use of a consensus standard.

Several commenters stated that the large number of modifications and limitations in the proposed rule is an indication that the NRC participation in the development of the ASME Code is not promoting the endorsement of the ASME Code in 10 CFR 50.55a as approved by the consensus process. The commenters emphasized that the NRC representatives participating in the ASME consensus process should voice concerns or propose alternative options, and cast negative votes when there are technical and regulatory concerns. This would allow other members on the committees to evaluate the NRC technical and regulatory concerns during the development of the Code, and thereby, reduce the number of modifications and limitations needed when incorporating the ASME Code by reference in 10 CFR 50.55a. The commenters also stated that Code changes are based on more than 30 years of plant operations and experience, years of research into better ways to inspect components or evaluate the results of inspection results, or the use of risk insights.

The NRC imposes limitations or modifications on the use of the consensus standards that are used in its regulatory process when the consensus standard does not adequately address a specific regulatory issue, the standard is technically incorrect, or it is inconsistent with current regulations. In accordance with NRC internal procedures, NRC representatives on ASME committees coordinate with other NRC to ensure that the views of NRC representatives on ASME committees are consistent with the views of the NRC. This coordination minimizes the need for modifications and limitations and, thus, reduces unnecessary regulatory burden. The NRC strives to develop technical positions in a timely manner for use in the standards development process. However, in instances when it is not practical for NRC to develop a position on an issue prior to casting its vote, NRC representatives on ASME committees are authorized to use their best judgement based on their experience, technical expertise, and discussion with other NRC staff. The goal that the NRC develop a final technical position on every Code change prior to voting on the change on the Main Committee level is not always achievable because of higher priority activities and current NRC staffing levels.

The NRC reviewed approximately 448 non-editorial Code changes during the rulemaking process to incorporate by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III and Section XI of the ASME BPV Code and the ASME OM Code. Although it may appear that there are a significant number of modifications and limitations in the final rule, limitations or modifications were imposed on a small fraction of the ASME Code non-editorial changes published in 1997 through 2000. Approximately 165 of the 448 non-editorial changes reviewed were considered reductions of Code requirements, and the NRC approved all but a small fraction of these non-editorial changes. In conclusion, the NRC finds the concern that the NRC participation in the development of the ASME Code is not promoting the endorsement of the ASME Code as approved by the consensus process, is not justified.

Comments on Backfit Requirements for Modifications and Limitations

The NRC is not imposing or mandating any new requirements in the limitations and modifications to Code provisions. It most instances, where limitations and modifications are imposed, the NRC requires the use of provisions of the ASME Code that have been previously approved. This is the case when those provisions have been unacceptably changed in later ASME Code editions and addenda. Several modifications restrict the use of a new Code provision while allowing a relaxation in the use of an earlier Code provision.

A number of commenters stated that the NRC imposition of exceptions (i.e., modifications and limitations) to the ASME Code are backfits and should be analyzed in accordance with the regulations in 10 CFR 50.109. To the contrary, the NRC finds that many of the modifications and limitations imposed during previous routine updates of 10 CFR 50.55a have not been considered backfits. The final rule dated August 6, 1992 (57 FR 34666), incorporated by reference in 10 CFR 50.55a the 1986 Addenda through the 1989 Edition of Section III and Section XI of the ASME BPV

Code. The backfit analysis section of the final rule (57 FR 34672) stated that a modification that simply retains an existing Section XI requirement is not a backfit. The final rule also added a requirement to expedite the implementation of the revised reactor vessel shell weld examinations in the 1989 Edition of Section XI. Imposing these examinations was considered a backfit because licensees were required to implement the examinations prior to the next 120-month ISI program inspection interval update.

The final rule dated August 8, 1996 (61 FR 41303), incorporated by reference in 10 CFR 50.55a the 1992 Edition with the 1992 Addenda of IWE and IWL of Section XI to require that containments be routinely inspected to detect defects that could compromise a containment's structural integrity. This action was considered a backfit because the Commission endorsed new subsections of the Code that expanded the scope of 10 CFR 50.55a to include components that were not considered by the existing regulations to be within the scope of ISI. The final rule dated September 22, 1999 (64 FR 51370), incorporated by reference in 10 CFR 50.55a the 1989 Addenda through the 1996 Addenda of Section III and Section XI of the ASME BPV Code, and the 1995 Edition with the 1996 Addenda of the ASME OM Code. The final rule expedited the implementation of the 1995 Edition with the 1996 Addenda of Appendix VIII of Section XI for qualification of personnel and procedures for performing UT examinations. The expedited implementation of Appendix VIII was considered a backfit because licensees were required to implement the new requirements in Appendix VIII prior to the next 120-month ISI program inspection interval update. The final rule also imposed modifications and limitations that retained existing ASME Code requirements that were not considered by the NRC to be backfits. In conclusion, modifications and limitations have historically not been considered to be backfits unless they expand the scope of the Code to include components that were not considered to be within the scope of ISI, or expedite the implementation of new Code provisions.

Limitations are also used to restrict the use of a new Code provision while expanding the use of an earlier Code provision. For example, 10 CFR 50.55a(b)(3)(vi) in the proposed rule prohibits the extension of the exercise interval for manual valves from 3 months (existing Code provision) to 5 years (new Code provision). 10 CFR 50.55a(b)(3)(vi) requires that manual valves be exercised every 2 years. In resolving this issue, the NRC could have retained the existing Code requirement to exercise manual valves every 3 months. However, the intent of the ASME consensus process was to extend the exercise interval for manual valves, and in this case, the NRC is accommodating the ASME consensus process to the extent that the NRC believes the extended exercise interval to 2 years is justified.

In conclusion, modifications and limitations are not considered backfits because they either retain existing Code provisions that have been previously approved by the NRC, or are a compromise between new and old Code provisions. Furthermore, the final rules dated September 22, 1999 (64 FR 51370), August 8, 1996 (61 FR 41303), and August 6, 1992 (57 FR 34666), were reviewed by the NRC's Committee to Review Generic Requirements prior to publication to ensure that backfits are identified and dispositioned in accordance with the requirements in 10 CFR 50.109.

II. Incorporation By Reference of "Code Case" Regulatory Guides

The NRC is proposing to revise its approach for approving ASME Code cases in order to fully satisfy the Administrative Procedures Act (APA) (5 USC 553) and 1 CFR Part 51, "Incorporation by Reference." The NRC is proposing to amend NRC regulations in 10 CFR 50.55a to incorporate by reference the NRC's regulatory guides (RGs) that address the use of Code cases prepared for the ASME BPV Code and OM Code. These "Code Case" regulatory guides currently are designated as RG 1.84, 1.85, and 1.147.

To date the NRC practice has been to review ASME BPV Code cases, assess the acceptability of each, and issue regulatory guides providing its conclusions on the acceptability of the Code cases. The NRC has referenced these RGs in Footnote 6 of 10 CFR 50.55a. Footnote 6 reads as follows:

> ASME Code cases that have been determined suitable for use by the Commission are listed in NRC Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability–ASME Section III Division 1," NRC Regulatory Guide 1.85, "Materials Code Case Acceptability–ASME Section III Division 1," and 1.147, "Inservice Inspection Code Case Acceptability–ASME Section XI Division 1." The use of other Code cases may authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to §50.55a(a)(3).

Recently, it has come to the NRC's attention that specific incorporation by reference by the Office of Federal Register (OFR) of these RGs has not previously been approved as required by 1 CFR Part 51. The NRC deemed many of the Code cases listed in these RGs acceptable (some with limitations) for licensees to implement as alternatives to the requirements in the ASME BVP Code. The NRC has found some Code cases unacceptable and has noted their unacceptability in the RGs. The NRC revises these RGs as new Code cases are published. Additionally, the reference to RGs in Footnote 6 does not give revision numbers of the RGs as also required by 1 CFR Part 51.

Furthermore, the NRC incorporates by reference various portions of the ASME BPV and OM Code requirements in 10 CFR 50.55a. Because these Code cases are usually alternatives to ASME Code requirements and not interpretations of how the requirements may be met, it is not permissible to use the RG process to approve licensee implementation of alternatives to these requirements. The approval to use these Code cases must be granted on a plant-specific basis or through rulemaking. Although the RGs are issued for public comment, general reference to the RGs addressing the ASME Code Cases in Footnote 6 of 10 CFR 50.55a could be viewed as contrary to the requirements of the APA, which requires that the public be given the opportunity to review, comment, and receive appropriate consideration of their comments prior to the imposition of Federal regulations.

The NRC held many internal discussions on this matter in order to reach a decision on how to endorse ASME Code cases in the most efficient and effective manner that met Federal procedural requirements. The NRC also held public meetings with external stakeholders to discuss the issue and obtain feedback on various approaches. As a result of these many discussions, the NRC concluded that the most effective and efficient approach for permitting licensees to use Code cases as alternatives to ASME Code requirements would be to incorporate by reference the RGs that list acceptable, conditionally acceptable, and unacceptable Code cases into 10 CFR 50.55a.

This would give the Code cases the same legal status as the portions of the ASME Code that are currently incorporated by reference in 10 CFR 50.55a. The approach would be accomplished through rulemaking by making the following revisions to 10 CFR 50.55a:

1. A new paragraph, 50.55a(i), containing the language of incorporation by reference would be added to 10 CFR 50.55a. This paragraph would identify each Code case RG by title and revision number.

2. Footnote 6 would be removed in its entirety. Note that Footnote 6 also contains the statement that the use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation. However, this provision is also contained in 10 CFR 50.55a(a)(3). Thus, its deletion from Footnote 6 will have no impact.

3. There are currently 12 references to Footnote 6 in 10 CFR 50.55a. Because each footnote reference would be deleted, a cross-reference to the appropriate portion of proposed paragraph (i) would be added with a statement that pursuant to 10 CFR 50.55a(i), licensees may use the Code cases that the NRC has found to be acceptable or conditionally acceptable as alternatives to the provisions in the ASME Codes.

Adopting this approach would establish a process of periodic rulemakings to incorporate by reference the latest regulatory guides which list all acceptable, conditionally acceptable, and unacceptable ASME Code cases in 10 CFR 50.55a. This approach would provide a sound regulatory basis for NRC's approval of the generic use of Code cases by licensees as alternatives to the provisions of the ASME Codes as incorporated by reference in NRC's regulations. Based on consultations with officials from the OFR, this approach would meet OFR requirements for incorporation by reference of documents in the regulations. The change in the Code case approval process will be seamless to licensees and would retain a process with which licensees are already familiar.

In addition, this approach would meet NRC's performance goal of maintaining safety by continuing to provide NRC review and approval of new ASME Code cases. It would reduce unnecessary regulatory burden by eliminating the need for licensees to submit plant-specific relief requests for NRC review and approval. It would also increase public confidence by allowing public participation in the process used to update the NRC's regulatory guides that approve, condition, or reject ASME Code cases as alternatives to the provisions of the ASME Code requirements.

The approach described above was discussed in SECY-01-0110, "Initiation of NRR-Sponsored Rulemaking: ASME BPV and OM Code Cases," dated June 21, 2001. The Commission approved the NRC's recommended approach in a staff requirements memorandum dated July 6, 2001. The proposed rule was issued on March 19, 2002 (67 FR 12488).

In summary, the NRC believes that this approach is a reasonable and legally sound approach that will eliminate the litigious risks associated with the existing approach. This option is responsive to the industry's desire for generic approval of ASME Code cases and is consistent with NRC's performance goals in that it maintains safety, makes more efficient use of NRC's and licensee's resources by eliminating the need for plant-specific reviews, and provides an opportunity for public involvement.

Revisions to NRC's Code Case Regulatory Guides

In conjunction with the Footnote 6 rulemaking described above, the NRC is preparing its next revisions to RGs 1.84, 1.85, and 1.147. There are several major changes to these RGs and approvals of significant, new Code cases that will appear in these next revisions and are worth mentioning.

The first major change is the combining of RG 1.84 (Section III design and fabrication) with RG 1.85 (Section III materials). Beginning with Revision 32, all Section III nuclear component Code cases that have been approved for use by the NRC will be listed in one regulatory guide. For this revision (32), the NRC reviewed Section III Code cases listed in Supplement 4 to the 1992 Edition through those listed in Supplement 10 to the 1998 Edition (except for those Code cases related to elevated-temperature, gas-cooled and liquid-metal reactors; Section III Division 2 components; and submerged spent fuel waste casks). This will be accomplished by placing all Section III design, fabrication, and materials Code cases into RG 1.84. It should be noted that RG 1.85 will no longer be updated, but it will not be withdrawn at this time because some Code cases contained in RG 1.85 continue to be used by licensees. The title of RG 1.84 will be changed to reflect the scopes of both RGs ("Design, Fabrication, and Materials Code Case Acceptability–ASME Section III, Division 1").

There are no major changes to RG 1.147 (Section XI ISI) other than to update the list of Code cases to include the latest ASME Code, Section XI ISI Code cases.

The second major change to the Code Case RGs is the introduction of a new (draft) regulatory guide addressing OM Code case acceptability. Draft Regulatory Guide DG-1089, "Operation and Maintenance Code Case Acceptability–ASME OM Code," is the first time that OM Code cases will be endorsed in a regulatory guide. The need for an OM Code case RG became apparent to the NRC when the NRC incorporated by reference for the first time the OM Code in a final rulemaking issued on September 22, 1999 (64 FR 51370). OM Code Cases OMN-1 through OMN-13 were reviewed for inclusion in this draft RG.

The Code Case RGs were issued for public comment on December 28, 2001 (66 FR 67335). The major changes to the Code Case RGs discussed above are summarized in Table 1 below.

Table 1 - Summary of Changes to Code Case Regulatory Guides

ASME Code Cases	NRC's Approval Document	
	Current	Proposed
Section III	RG 1.84 (design and fabrication) RG 1.85 (materials)	RG 1.84 (design, fabrication, and materials) Rev. 32
Section XI	RG 1.147 (ISI)	RG 1.147 (ISI) Rev.13
OM Code	none	new RG (draft DG-1089)

It should be noted that many of the OM Code cases approved by the NRC in the draft RG implement risk-informed alternatives to IST requirements for pumps and valves. These Code cases may be used by licensees (when the RG is issued in final form) without a need to request NRC review and approval provided they are used with any conditions as identified in the final RG. With the incorporation by reference of the OM Code Case RG (draft DG-1089), if a licensee voluntarily elects to use the Code Case, the conditions specified in the RG are regulatory requirements, not guidance or recommendations.

OM Code cases that have not yet been reviewed and approved by the NRC in the draft RG may be implemented pursuant to 10 CFR 50.55a(a)(3) which permits the use of alternatives to the regulations in §50.55a provided that the proposed alternative can be demonstrated to provide an acceptable level of quality and safety and its use is authorized by NRC's Director of the Office of Nuclear Reactor Regulation.

III. Conclusion

The final rule to update 10 CFR 50.55a to incorporate by reference a more recent edition and addenda to the ASME OM Code is scheduled to be issued in September 2002. The next update to 10 CFR 50.55a will incorporate by reference the 2001 Edition, 2002 Addenda, and 2003 Addenda of the Section III, Division 1, and Section XI of the ASME BPV Codes and the ASME OM Code. The final rule will become effective 60 days from date of publication in the *Federal Register*. The final rule to amend the regulations in 10 CFR 50.55a to incorporate by reference the NRC's RGs that address the use of Code Cases prepared for the ASME BPV Code and OM Code is scheduled to be issued in March 2003. The next revision to Code Case RGs 1.84, 1.85, and 1.147 are scheduled to be issued at the same time as the final rule.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The 2002 Symposium on Valve and Pump Testing, jointly sponsored by the Board on Nuclear Codes and Standards of the American Society of Mechanical Engineers and by the U.S. Nuclear Regulatory Com mission, provides a forum for exchanging information on technical and regulatory issues associated with the testing of v alves and pumps used in nuclear power plants. The symposium provides an opportunity to discuss the need to improve that testi ng to help ensure the reliable performance of valves and pumps. The participation of industry representatives, regulatory pe rsonnel, and consultants ensures the discussion of a broad spectrum of ideas and perspectives regarding the improvement of test ing programs and methods at nuclear power plants.

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