March 24, 2000

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

Dear Mr. Tolstykh:

Enclosed are the following IRS reports:

- RESOLUTION OF GENERIC SAFETY ISSUE 158: PERFORMANCE OF SAFETY-RELATED POWER-OPERATED VALVES UNDER DESIGN BASIS CONDITIONS (NRC Regulatory Issue Summary 2000-03).

- RESOLUTION OF GENERIC SAFETY ISSUE 165, SPRING-ACTUATED SAFETY AND RELIEF VALVE RELIABILITY (NRC Regulatory Issue Summary 2000-05).

Each report is being submitted in the following two media: (1) a hard copy of the input file for the AIRS database; and (2) a 3.5-inch HD diskette containing the input file for the AIRS database in Microsoft Word 6.0 format.

If you have any questions regarding these reports, please call Eric J. Benner of my staff. He can be reached at (301) 415-1171.

Sincerely,

/RA/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications and
  Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enclosures 1 and 2:
Mr. Lennart Carlsson
Nuclear Safety Division
Nuclear Energy Agency
Organization for Economic
  Cooperation and Development
Le Seine Saint Germain
12, Boulevard des Iles

92130, Issy-les-Moulineaux, France

March 24, 2000

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/03/15	

EVENT TITLE

RESOLUTION OF GENERIC SAFETY ISSUE 158: PERFORMANCE OF SAFETY-RELATED POWER-OPERATED VALVES UNDER DESIGN BASIS CONDITIONS
(NRC Regulatory Issue Summary 2000-03)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses the closure of Generic Safety Issue (GSI) 158, “Performance of Safety- Related Power-Operated Valves Under Design Basis Conditions,” and of the NRC staff’s intent to continue to work with industry groups and to monitor addressees’ activities to ensure that safety-related power-operated valves (POVs) are capable of performing their specified functions under design basis conditions. GSI-158, “Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,” was identified by the NRC after reactor operating experience and research results on motor-operated valves (MOVs), solenoid-operated valves (SOVs), air-operated valves (AOVs), and hydraulically operated valves (HOVs) indicated that testing under static conditions was insufficient to demonstrate consistent performance of these valves under design-basis conditions.

RESOLUTION OF GENERIC SAFETY ISSUE 158: PERFORMANCE OF SAFETY-RELATED POWER-OPERATED VALVES UNDER DESIGN BASIS CONDITIONS
(NRC Regulatory Issue Summary 2000-03)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	**1.4**		
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.AB**	**3.BE**	**3.DB**
4.	**Failed/Affected Components:**	**4.2.3**		
5.	**Cause of the Event:**	**5.3.1**		
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.0**		
8.	**Nature of Failure or Error:**	**8.3**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 15, 2000

NRC REGULATORY ISSUE SUMMARY 2000-03
RESOLUTION OF GENERIC SAFETY ISSUE 158: PERFORMANCE OF SAFETY-RELATED POWER-OPERATED VALVES UNDER DESIGN BASIS CONDITIONS

Addressees

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Intent

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to inform addressees of the closure of Generic Safety Issue (GSI) 158, “Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,” and of the staff’s intent to continue to work with industry groups and to monitor addressees’ activities to ensure that safety-related power-operated valves (POVs) are capable of performing their specified functions under design basis conditions. No action or written response is requested.

Background Information

GSI-158, “Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,” was identified by the NRC after reactor operating experience and research results on motor-operated valves (MOVs), solenoid-operated valves (SOVs), air-operated valves (AOVs), and hydraulically operated valves (HOVs) indicated that testing under static conditions was insufficient to demonstrate consistent performance of these valves under design-basis conditions. Operating events involving observed or potential common-cause failures were documented in NUREG-1275, “Operating Experience Feedback Report,” Volumes 2 and 6 for air systems and SOVs, respectively, and in AEOD/C603, “Review of Motor-Operated Valve Performance,” for MOVs. These issues are also more recently discussed in NUREG/CR-6644, “Generic Issue 158: Performance of Safety-Related Power-Operated Valves Under Operating Conditions.” Two related documents, NUREG-1275, Vol.13, "Evaluation of Air-Operated Valves at U.S. Light-Water Reactors," and NUREG/CR-6654, "A Study of Air-Operated Valves in U.S. Nuclear Power Plants," are focused specifically on AOVs.

MOV performance issues were not discussed in GSI-158, as they were documented previously in the resolution of TMI Action Plan Item II.E.6.1, “In Situ Testing of Valves-Test Adequacy Study,” which is included in NUREG-0933, “Prioritization of Generic Safety Issues.” Resolution of this issue resulted in the issuance of Generic Letter (GL) 89-10, “Safety-Related

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Motor-Operated Valve Testing and Surveillance," and subsequently GL 96-05, "Periodic Verification of Design Basis Capability of Safety-Related Motor-Operated Valves."

The NRC staff briefed the Advisory Committee on Reactor Safeguards (ACRS) on May 6, 1999, and presented the results of staff and contractor studies of the POV issue. The staff concluded that no new regulations were required to address the issues contained in GSI-158. The current regulations provide an adequate framework to address specific POV issues similar to MOV issues that were resolved through the issuance of GL 89-10 and GL 96-05. The staff also described the voluntary industry initiative being developed to address AOV issues. In its May 14, 1999, letter to the NRC staff, the ACRS stated that the central issue, whether POVs are able to perform their intended functions under design basis dynamic conditions, had not been adequately addressed. The ACRS further stated that unless the NRC staff undertakes a pro-active effort to ensure resolution of this issue, the industry initiative will remain an optional, voluntary program that will not fully resolve the concerns of GSI-158.

Most of the recent staff and industry attention has been focused on AOV performance. The NRC staff stated in a July 2, 1999, letter to the ACRS that it would continue to monitor and work with industry groups developing design basis verification and testing programs for AOVs. The NRC staff noted further that if the actions of the industry did not adequately address the functionality of POVs under design basis dynamic conditions, the NRC staff would take additional regulatory action as appropriate.

The NRC staff previously requested that the industry verify the capability of AOVs with respect to issues involving the plant instrument air supply system. In GL 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," addressees were requested to verify by test that air-operated safety-related components will perform as expected in accordance with all design-basis events. All addressees were required to respond to the generic letter with confirmation that this verification had been performed. All responses were received by 1993 and the generic letter was subsequently closed.

<u>Recent AOV Performance and Safety Significance</u>

A recent NRC study of AOVs, documented in NUREG-1275, Vol.13, and NUREG/CR-6654, included a review of AOV operating experience and the results of 7 site visits to 11 U.S. light water reactors conducted in 1997-1998. The seven licensees collectively identified a total of 167 safety-related, high-risk-significant AOVs, ranging from an individual reactor facility high of 36 AOVs to a low of 4 AOVs. In addition, two of those licensees identified a total of 15 AOVs that were non safety-related but high-risk-significant. Most of the licensees visited were planning to verify the design-basis capability of all the referenced AOVs. The licensees' determinations of the high-risk-significant AOVs were based on a variety of methods, including plant-specific probabilistic risk assessment, individual plant examination, and maintenance rule expert panel reviews. Many of the licensees' determinations included evaluations of the risk achievement worth (RAW) and Fussell-Vesely (F-V) risk rankings of the AOVs. Each categorization method was unique.

The major safety concern identified in the NRC AOV study from a risk perspective is the simultaneous common-cause failure of AOVs which disable redundant trains of a system important to safety. The scenario of most concern is that during an accident or transient, AOVs in redundant trains of a safety system fail when subjected to pressure, temperature, and flow conditions different from those seen during normal operation or testing. As discussed in the NRC AOV study, some licensees found that certain AOVs had high RAW and/or F-V risk rankings. Table 6 of NUREG/CR-6654 includes the RAW values for AOVs that were calculated by licensees at three plants. These calculations showed that, in some cases, the RAW could increase by one or two orders of magnitude as a result of common-cause failures. RAW for common-cause AOV failures at those three plants ranged from slightly over 1 to 202. Weaknesses in the design, testing, and maintenance of AOVs could result in common-cause AOV failures which are not addressed in plant safety analyses. The common-cause AOV failures that have been documented in the AOV study did not occur simultaneously with design-basis transients but were identified during operations, maintenance, or testing.

Industry Initiatives

The Joint Owners Group on Air Operated Valves (JOG AOV), which is facilitated by the Nuclear Energy Institute (NEI), presented to the NRC staff in a public meeting on June 3, 1999, the industry's voluntary program to address AOV issues. The JOG AOV program provides guidance to verify valve performance at design conditions and long-term periodic verification of safety-related AOVs categorized as high-risk-significant. For safety-related, low-risk-significant AOVs and AOVs that are not safety-related but are determined to be high-risk-significant, the JOG AOV program also provides guidance for a less-rigorous verification of valve functionality. The activities for safety-related, low-risk-significant AOVs and non safety-related, high-risk-significant AOVs would not necessarily involve verification that the valves would perform under design conditions or require long-term periodic verification. The methodology to determine valve safety significance, as specified in the industry program, may include such risk insight methods as in Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” or programs established to meet the requirements of 10 CFR 50.65, “Requirements for monitoring the effectiveness of maintenance at nuclear power plants”, (the maintenance rule) in combination with individual plant examinations and the review performed by a separate expert panel.

This industry program document was completed and distributed to utilities in 1999. The NRC received a copy of the program document in a letter from D. Modeen (NEI) to E. Imbro (NRC), dated July 19, 1999. NRC comments on the JOG AOV program and its implementation were sent to NEI in a letter from E. Imbro to D. Modeen, dated October 8, 1999. Although the program was noted to have some limitations, the NRC staff recognizes that industry-wide implementation of this program would achieve a uniform level of consistency that would provide increased confidence in the design-basis capabilities of high-risk-significant AOVs in nuclear power plants. As stated above, the NRC will continue to work with industry groups to ensure that safety-related POVs are capable of performing their specified functions under design basis conditions. If POV functionality under design basis conditions is not adequately addressed by the industry, the NRC staff will take additional regulatory action as appropriate.

Summary of the Issue

The NRC has closed GSI-158 on the basis that current regulations provide adequate requirements to ensure verification of the design-basis capability of POVs and that no new regulatory requirements are needed. The NRC staff will continue to work with industry groups on an industry-wide approach to the POV issue and to provide timely, effective, and efficient resolution of the concerns regarding POV performance. The NRC staff will also continue to monitor licensees' activities to ensure that POVs are capable of performing their specified safety-related functions under design-basis conditions.

Voluntary Initiatives

Although there are no regulatory requirements for licensees to establish an AOV program, licensees are required by 10 CFR 50.65 to monitor the performance of structures, systems, or components (SSCs) in a manner sufficient to provide reasonable assurance that such SSCs (e.g., systems with safety-related and high-safety-significant AOVs) are capable of fulfilling their intended functions. Addressees who implement the JOG AOV program to help ensure the design basis capability of AOVs may wish to consider the NRC comments contained in the NRC's letter to NEI dated October 8, 1999. Addressees who choose to develop plant-specific AOV programs may wish to consider the attributes listed in the attachment to this RIS. These attributes are based on lessons learned from the staff's involvement in the activities related to GL 89-10 and the NRC site visits documented in NUREG/CR-6654.

Backfit Discussion

This RIS requests no action or written response. Therefore, this RIS is not a backfit under 10 CFR 50.109, and the staff did not perform a backfit analysis.

Federal Register Notification

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational and the NRC staff discussed the closure of GSI-158 in a public meeting with the ACRS.

If you have any questions about this RIS, please contact one of the technical contacts listed below.

/RA/S. F. Newberry FOR

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Attachment:
1. Successful Power-Operated Valve Program

Technical Contacts: Joseph Colaccino, NRR
301-415-2753
E-mail: jxc1@nrc.gov

Harold L. Ornstein, RES
301-415-7574
E-mail: hlo@nrc.gov

ATTRIBUTES OF A SUCCESSFUL POWER-OPERATED VALVE DESIGN CAPABILITY AND LONG-TERM PERIODIC VERIFICATION PROGRAM

1. Include all maintenance rule scope power-operated valves (POVs) in program.

2. Verify POVs in non-safety position are capable of returning to their safety position if train is assumed operable with valves in their non-safety position.

3. For air-operated valves, verify guidance in GL 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," has been successfully implemented, including periodic monitoring of air quality.

4. Evaluate motor-operated valve (MOV) risk-ranking methodologies developed by the Boiling Water Reactor Owners Group and the Westinghouse Owners Group for applicability to risk ranking of POVs at the specific plant, as applicable.

5. Focus initial efforts on safety-related, active, high-risk POVs. Information obtained from these valves and lessons learned may be used to verify and maintain design basis capability of similar safety-related POVs.

6. Verify methods for predicting POV operating requirements using MOV lessons learned or specific POV dynamic diagnostic testing. Use of the Electric Power Research Institute (EPRI) MOV Performance Prediction Method must include all guideline aspects of that methodology and not only individual EPRI valve test results.

7. Justify method for predicting POV actuator output capability by test-based program established by the vendor, licensee, or industry.

8. Address all applicable weak links, including actuator, valve, and stem.

9. Ensure quality assurance program coverage.

10. Provide sufficient diagnostics when baseline testing to verify capability. Diagnostics might not be needed if normal plant operation frequently demonstrates design basis capability.

11. Specify when dynamic or static diagnostic periodic testing is needed.

12. Ensure post-maintenance testing is adequate to verify capability of all safety-related POVs and risk-significant functions of non-safety-related POVs.

13. Ensure POV maintenance procedures are reviewed to incorporate lessons learned from other valve programs.

14. Upgrade training to incorporate lessons learned from other valve programs.

15. Apply feedback from plant-specific and industry information, including test data, to all applicable safety-related POVs.

16. Establish quantitative (test data) and qualitative (maintenance and condition reports) trending of POV performance with detailed review following each refueling outage.

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/03/16	

EVENT TITLE

RESOLUTION OF GENERIC SAFETY ISSUE 165, SPRING-ACTUATED SAFETY AND RELIEF VALVE RELIABILITY (NRC Regulatory Issue Summary 2000-05)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses the NRC staff's resolution of Generic Safety Issue (GSI) 165, "Spring-Actuated Safety and Relief Valve Reliability," and provides information that may help licensees determine possible improvements in configuring and/or operating various plant systems to reduce safety and relief valve (SRV) malfunctions. Spring-actuated SRVs are used in many systems to protect piping and components from potential overpressure conditions. GSI 165 was identified after failure of a spring-actuated SRV at the Shearon Harris nuclear plant degraded the high-head safety injection (HHSI) system and went undetected for a significant period. The concerns that led to the development of the GSI were as follows: (1) the failure was serious, (2) there were no American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements for testing most SRVs, and (3) little attention had been focused on these components. Consequently, it could not be concluded that the failure at Shearon Harris was unique.

RESOLUTION OF GENERIC SAFETY ISSUE 165, SPRING-ACTUATED SAFETY AND RELIEF VALVE RELIABILITY (NRC Regulatory Issue Summary 2000-05)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	**1.4**		
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.AF**	**3.BF**	**3.BH**
4.	**Failed/Affected Components:**	**4.2.3**		
5.	**Cause of the Event:**	**5.1.0**		
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.0**		
8.	**Nature of Failure or Error:**	**8.3**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

March 16, 2000

NRC REGULATORY ISSUE SUMMARY 2000-05 RESOLUTION OF GENERIC SAFETY ISSUE 165, SPRING-ACTUATED SAFETY AND RELIEF VALVE RELIABILITY

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those licensees who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to notify nuclear power reactor licensees about the staff's resolution of Generic Safety Issue (GSI) 165, "Spring-Actuated Safety and Relief Valve Reliability," and to provide information that may help licensees determine possible improvements in configuring and/or operating various plant systems to reduce safety and relief valve (SRV) malfunctions. This RIS does not transmit any new requirements or staff positions. No specific action or written response is required.

BACKGROUND INFORMATION

Spring-actuated SRVs are used in many systems to protect piping and components from potential overpressure conditions. GSI 165 was identified after failure of a spring-actuated SRV at the Shearon Harris nuclear plant degraded the high-head safety injection (HHSI) system and went undetected for a significant period. The concerns that led to the development of the GSI were as follows: (1) the failure was serious, (2) there were no American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) requirements for testing most SRVs, and (3) little attention had been focused on these components. Consequently, it could not be concluded that the failure at Shearon Harris was unique. The combination of the operating conditions of the HHSI SRVs, the normally cross-tied configuration of the system trains, and the difficulty of detecting the failures, resulted in a conditional core damage probability of 6×10^{-3} per reactor-year (RY) at the Shearon Harris plant, that could be attributed to failure of the HHSI SRVs to open or close as required. In its initial assessment, the staff used a conservatively bounding methodology in which it was estimated that up to 8 percent of the approximately 60 valves of this type in various safety-related systems in a typical plant could be significant contributors to core damage frequency (CDF), if they failed. In addition, the staff estimated that 10 percent of the SRVs would have the capability to fail their trains. A preliminary analysis

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indicated that failure of SRVs in a typical plant could raise the CDF to an upper bound value of 5×10^{-2}/RY. Finally, the staff estimated that the SRV failure probability could be significantly reduced by having licensees perform economical tests in their plants.

Because significant NRC and industry resources had been spent on both evaluating the risk and improving the reliability of pressurizer safety valves and main steam safety valves in pressurized-water reactors and main steam SRVs in boiling-water reactors, the focus of this issue was limited to smaller spring-actuated SRVs in safety-related support systems and the effects of their unreliability on plant operation.

SUMMARY OF ISSUE

In its evaluation and resolution of GSI 165, the staff found that its initial assumptions used in estimating the risk significance of the issue were overly conservative or had changed. Review of piping and instrument diagrams for important safety-related systems revealed that Shearon Harris was somewhat unique and that the type of system cross-tying that contributed to the seriousness of the Shearon Harris degradation was not present at most other plants. Review of related valve data identified only a single valve configuration found in high pressure safety injection (HPSI) systems in Combustion Engineering plants that had the potential for failing its train. That SRV configuration was analyzed as a worst case. Due to the low probability of failure of both trains of the system, the analysis showed that the increase in CDF for that SRV configuration was acceptably low, (6×10^{-6}/RY). Review of licensee event reports and Nuclear Plant Reliability Data System narratives during data gathering for failure rate estimation did not reveal any other instances of valve spring failure, other than the failure at Shearon Harris. Finally, and perhaps most significantly, the additional testing requirements for these SRVs contemplated in the initial assessment for the resolution of this GSI were included in the 1986 Edition of the ASME Code inservice testing (IST) requirements and continue to be required in the ASME Code. The additional testing is already being performed at all but a few plants, which still use earlier editions of the ASME Code (i.e., 1983 Edition). It is expected that all plants will have updated their IST programs to include the additional testing by the end of 2002. As a result, the NRC staff closed GSI 165 without any new regulatory requirements being issued.

Backfit Discussion

This RIS requests no action or written response. Consequently, the staff did not perform a backfit analysis.

Federal Register Notification

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational.

If there are any questions about this matter, please contact one of the persons listed below, or the appropriate Office of Nuclear Reactor Regulation project manager for a specific nuclear power plant.

/RA/

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

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