September 14, 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:

- STEAM GENERATOR TUBE FAILURE AT INDIAN POINT UNIT 2 (NRC Information Notice 2000-09).

- LICENSEE RESPONSIBILITY FOR QUALITY ASSURANCE OVERSIGHT OF CONTRACTOR ACTIVITIES REGARDING FABRICATION AND USE OF SPENT FUEL STORAGE CASK SYSTEMS (NRC Information Notice 2000-11).

- RESOLUTION OF GENERIC SAFETY ISSUE B-55, "IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES" (NRC Regulatory Issue Summary 2000-12).

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

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

September 14, 2000

Dear Mr. Tolstykh:

Enclosed are the following IRS reports:

- STEAM GENERATOR TUBE FAILURE AT INDIAN POINT UNIT 2 (NRC Information Notice 2000-09).
- LICENSEE RESPONSIBILITY FOR QUALITY ASSURANCE OVERSIGHT OF CONTRACTOR ACTIVITIES REGARDING FABRICATION AND USE OF SPENT FUEL STORAGE CASK SYSTEMS (NRC Information Notice 2000-11).
- RESOLUTION OF GENERIC SAFETY ISSUE B-55, “IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES” (NRC Regulatory Issue Summary 2000-12).

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

Distribution (Transmittal Letter, Cover Sheet, and Coding Sheet Only):

REXB R/F	SCollins, NRR	MCullingford, NRR	JDunn Lee, OIP
FMiraglia, DEDE	MSatorius, OEDO	JLieberman, OIP	PUBLIC

Accession No. ML003749427

To receive a copy of this document, Indicate in the box "C" copy w/o attach/encls "E" copy w/attach/encls "N" No copy

OFFICE	REXB/DRIP	E	REXB/DRIP	N	C:REXB/DRIP	E
NAME	EBenner		KGray		LMarsh	
DATE	09/14/2000		09/14/2000		09/14/2000	

OFFICIAL RECORD COPY

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/06/28	

EVENT TITLE

STEAM GENERATOR TUBE FAILURE AT INDIAN POINT UNIT 2
(NRC Information Notice 2000-09)

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 a steam generator tube failure which occurred at the Indian Point Unit 2 nuclear plant on February 15, 2000, at 7:17 p.m., which required the declaration of an Alert at 7:29 p.m., and a manual reactor trip at 7:30 p.m. The operators identified that the #24 steam generator was the source of the leak and completed isolation of the #24 steam generator by 8:31 p.m. The event was risk significant. It involved a steam generator tube failure that resulted in an initial primary-to-secondary leak of reactor coolant of approximately 146 gallons per minute and required an “Alert” declaration (the second level of emergency action in the NRC-required emergency response plan). The event resulted in a minor radiological release to the environment that was well within regulatory limits. No radioactivity was measured offsite above normal background levels, and the event did not adversely impact the public health and safety.

STEAM GENERATOR TUBE FAILURE AT INDIAN POINT UNIT 2
(NRC Information Notice 2000-09)

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

1.	Reporting Categories:	1.2.2	1.3.4	1.4
2.	Plant Status Prior to the Event:	2.1.1		
3.	Failed/Affected Systems:	3.AH		
4.	Failed/Affected Components:	4.2.6		
5.	Cause of the Event:	5.1.1.7		
6.	Effects on Operation:	6.1.2	6.4	6.5.2
7.	Characteristics of the Incident:	7.2		
8.	Nature of Failure or Error:	8.2		
9.	Nature of Recovery Actions:	9.1.1		

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

June 28, 2000

NRC INFORMATION NOTICE 2000-09: STEAM GENERATOR TUBE FAILURE AT INDIAN POINT UNIT 2

Addressees

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

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of a steam generator tube failure at Indian Point Unit 2. NRC investigations of the licensee's steam generator inspection program are ongoing and any potentially generic issues identified will be communicated in a separate generic communication. However, the investigations to date re-emphasize the importance of licensee involvement with ongoing industry efforts to understand and detect steam generator degradation. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

On February 15, 2000, at 7:17 p.m., the Indian Point Unit 2 nuclear plant experienced a steam generator tube failure, which required the declaration of an Alert at 7:29 p.m., and a manual reactor trip at 7:30 p.m. The operators identified that the #24 steam generator was the source of the leak and completed isolation of the #24 steam generator by 8:31 p.m.

At 9:02 p.m., the operator opened the high-pressure steam dump valves and established an excessive primary plant cooldown rate that caused a rapid reduction in the pressurizer level and required the operators to manually initiate safety injection. The operators reset the safety injection at 9:21 p.m., reduced the reactor coolant system pressure to about 970 psig at 9:32 p.m., and re-commenced a plant cooldown at 11:35 p.m.

The residual heat removal (RHR) system was placed in service on February 16, 2000, at 12:38 p.m., and primary plant pressure was reduced below the #24 steam generator pressure to terminate the steam generator tube leakage at 2:20 p.m. The plant cooldown continued, and the plant entered cold shutdown at 4:57 p.m. The licensee exited the Alert at 6:50 p.m.

The NRC sent an Augmented Inspection Team (AIT) on February 18, 2000, to review the causes, safety implications, and licensee actions associated with the event. The AIT developed

ML003726124

a sequence of events, determined the risk significance of the event, and assessed the response by the plant staff and management. The cause of the tube failure was outside the scope of this inspection and is currently being reviewed separately by the NRC. The AIT's report is presented in Inspection Report 05000247/2000-02, dated April 28, 2000 (Accession Number ML003710036).

Discussion

The event was risk significant. It involved a steam generator tube failure that resulted in an initial primary-to-secondary leak of reactor coolant of approximately 146 gallons per minute and required an "Alert" declaration (the second level of emergency action in the NRC-required emergency response plan). The event resulted in a minor radiological release to the environment that was well within regulatory limits. No radioactivity was measured offsite above normal background levels, and the event did not adversely impact the public health and safety.

The licensee performed the necessary actions to protect the health and safety of the public. Specifically, the operators promptly and appropriately took those actions in the emergency operating procedures to trip the reactor, isolate the affected steam generator, and depressurize the reactor coolant system. Additionally, the necessary event mitigation systems worked properly. Notwithstanding the above actions, the AIT identified performance problems in several broad areas that challenged operators, complicated the event response, delayed achieving the cold shutdown condition, and affected the radiological release. The problems involved operator performance, procedure quality, equipment performance, technical support, and emergency response.

Operator Performance

Some operator performance problems were noted during the plant cooldown phase involving the following:

- While attempting to cool down the reactor coolant system (RCS), the reactor operator initiated an excessive cooldown rate that exceeded procedural and Technical Specification limits. The excessive cooldown led to several conditions that complicated the subsequent event response and delayed the RCS cooldown.
- Operators were slow to recognize configuration lineup problems that (1) prevented successful operation of the auxiliary spray system to lower RCS pressure and (2) delayed heatup of the RHR system.

Procedure Quality

The procedures adequately guided the initial operator response; however, several procedure problems were identified that delayed the cooldown and depressurizing of the RCS. Procedure deficiencies affected Standard Operating Procedures, Emergency Operating Procedures, and Emergency Plan Implementing Procedures. Specific activities included initiation of RHR cooling, initiation of component cooling water alignment, use of auxiliary pressurizer spray, use of methods to monitor RCS temperature to maintain cold shutdown conditions, and initiation of

emergency response organization (ERO) notifications. Station personnel were previously aware of the procedure issue involving initiation of RHR cooling but had not corrected the problem before this event.

Equipment Performance

The necessary event mitigation systems, including the reactor protection system, the auxiliary feedwater system, and the safety injection system, functioned properly. However, several longstanding equipment performance problems were identified that challenged operators during this event:

- Two losses of condenser vacuum resulted from problems with the operation of the automatic steam supply pressure control valve to the steam jet air ejectors, and the #22 condenser vacuum pump.
- The isolation valve seal water system became inoperable during the event and required operator action and an entry into a Technical Specification Limiting Condition for Operation Action Statement.
- A containment entry was required to install a temporary nitrogen supply to the pressurizer power-operated relief valve to compensate for a design deficiency. This action was required before placing the overpressure protection system in service.
- The steam generator leak rate monitoring equipment had been degraded for an extended period, and limited the amount of steam generator leak rate information available to the operators before the event.
-

The AIT determined that the number and duration of the equipment problems reflected weaknesses in engineering, corrective action processes, and operational support at the station. The licensee's response to a number of the equipment problems identified during the event reflected an acceptance of "working around" the problem rather than fixing it.

Emergency Response

The ERO took the necessary steps to ensure the protection of public health and safety. The operators properly classified the event, and the licensee implemented a thorough peer review of the emergency response to this event. The AIT identified several emergency plan and implementing procedure problems similar to those identified by the licensee's peer review team, including the following:

- The emergency response staff was slow to activate the emergency facilities.
- The licensee was slow to establish accountability (i.e., identify the location) of emergency response personnel.

- The emergency response data system (ERDS) was inoperable for the first several hours of the event as a result of a pre-existing equipment problem.
- Problems were noted in the implementation of the media response plan.
- Problems were identified involving the timeliness and quality of technical support provided to the operators.

The licensee developed and was in the process of implementing an emergency response improvement plan before the event.

This information notice requires no specific action or written response. However, recipients are reminded that they are required to consider industry-wide operating experience (including NRC information notices) when practical when setting goals and performing periodic evaluations under Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," of Part 50 of Title 10 of the Code of Federal Regulations. If you have any questions about the information in this notice, please contact the one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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

Technical contacts:

Eric Benner, NRR
301-415-1171
E-mail: ejb1@nrc.gov

Lawrence Doerflein, Region I
610-337-5378
E-mail: ltd@nrc.gov

Peter Eselgroth, Region I
610-337-5234
E-mail: pwe@nrc.gov

Raymond Lorson, Region I
603-474-3589
E-mail: rkl@nrc.gov

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/08/07	

EVENT TITLE

LICENSEE RESPONSIBILITY FOR QUALITY ASSURANCE OVERSIGHT OF CONTRACTOR ACTIVITIES REGARDING FABRICATION AND USE OF SPENT FUEL STORAGE CASK SYSTEMS (NRC Information Notice 2000-11)

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 reminds United States general and site specific licensees of their responsibilities to assure that the quality assurance requirements of Part 72, Subpart G, to Title 10 of the U.S. Code of Federal Regulations (CFR) have been met before a dry cask storage system is placed in service at their nuclear power plants. The regulations require that nuclear power plant licensees assume full responsibility for the overall safety and operational use of the dry cask storage system at their sites. The nuclear power plant licensee is also responsible for assuring that the fabrication and preparation for use of the dry cask storage system, and the contractor's activities associated with the dry cask storage system, conform with NRC regulations, the Certificate of Compliance (CoC), and the license conditions for the nuclear power plant. This IN discusses a number of examples of inadequate implementation of quality assurance (QA) programs identified in recent NRC inspections.

LICENSEE RESPONSIBILITY FOR QUALITY ASSURANCE OVERSIGHT OF CONTRACTOR ACTIVITIES REGARDING FABRICATION AND USE OF SPENT FUEL STORAGE CASK SYSTEMS (NRC Information Notice 2000-11)

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.KG**		
4.	**Failed/Affected Components:**	**4.2.5**		
5.	**Cause of the Event:**	**5.1.1**		
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.0**		
8.	**Nature of Failure or Error:**	**8.1**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
OFFICE ON NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 7, 2000

NRC INFORMATION NOTICE 2000-11: LICENSEE RESPONSIBILITY FOR QUALITY ASSURANCE OVERSIGHT OF CONTRACTOR ACTIVITIES REGARDING FABRICATION AND USE OF SPENT FUEL STORAGE CASK SYSTEMS

Addressees:

All U.S. Nuclear Regulatory Commission (NRC) 10 CFR Part 50 and Part 72 licensees, and Part 72 Certificate of Compliance holders.

Purpose:

The NRC is issuing this information notice (IN) to remind general and site specific licensees of their responsibilities to assure that the quality assurance requirements of Part 72, Subpart G, to Title 10 of the U.S. Code of Federal Regulations (CFR) have been met before a dry cask storage system is placed in service at their nuclear power plants. The regulations require that nuclear power plant licensees assume full responsibility for the overall safety and operational use of the dry cask storage system at their sites. The nuclear power plant licensee is also responsible for assuring that the fabrication and preparation for use of the dry cask storage system, and the contractor's activities associated with the dry cask storage system, conform with NRC regulations, the Certificate of Compliance (CoC), and the license conditions for the nuclear power plant. This IN discusses a number of examples of inadequate implementation of quality assurance (QA) programs identified in recent NRC inspections. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this IN are not new NRC requirements; therefore, no specific action nor written response is required.

Description of Circumstances:

During an inspection in 1999, NRC staff identified cracks in the front opening of fabricated horizontal spent fuel storage modules that could have affected their structural integrity. The NRC further learned that the applicant's contractor and subcontractor had accepted these cracks without documenting them in either nonconformance reports (NCRs) or process deficiency reports. As a result, about 29 storage modules were found to have similar cracks. It was determined that the acceptance of this nonconforming condition was based on the contractor's reliance on inapplicable acceptance criteria, inadequate worker training, and a lack of understanding of the role of the NCR process. This could have been avoided if the applicant had been more actively and effectively involved in overseeing its contractor's quality assurance activities.

ML003734271

Problems in the implementation of the program by the applicant's contractor and subcontractor, occurred in part because of the lack of effective oversight. The identified deficiencies were corrected prior to the issuance of the Part 72 license.

During another inspection in 1999, the NRC reviewed the implementation of the licensee's QA program with respect to the fabrication of spent fuel storage casks. The licensee had contracted with a vendor for the delivery of pressurized water reactor (PWR) fuel baskets for use at the Independent Spent Fuel Storage Facility. The licensee had established a QA program that was found to meet regulatory requirements. However, inspectors found instances of inadequate oversight of contractor and subcontractor activities. Specifically, the inspection team identified a number of examples of violations of 10 CFR 72.154, "Control of purchased material, equipment, and services."

In one example, the NRC determined that the licensee did not effectively oversee the design control process when changes were made to the coating application process for the storage cask fuel basket internals. A number of design drawings required that the fuel basket protective coating be applied according to the manufacturer's specifications. This included spray application of coating followed by high temperature curing. The vendor instructed its contractor to brush-apply coatings to basket internal areas not adequately covered by the spray process, but the vendor failed to complete a design change/request notice to support and justify this change as required by its procedure. After being informed of this issue, the vendor initiated a corrective action report to resolve the matter.

Other examples of inadequate oversight of contractor's activities were identified by the inspection in the areas of design development, material procurement, fabrication and assembly, and test and inspection. Specifically, the contractor's fabrication procedure contained conflicting requirements for fabrication quality assurance. In addition, the contractor did not have a procedure in place for controlling access to its small parts storage area, personnel were not adequately controlling ink stamps used for documenting work acceptance on fabrication travelers, and personnel had performed dimensional acceptance measurements using uncalibrated measurement and test equipment.

Discussion:

Section 140, of 10 CFR Part 72, states that, "the certificate holder and applicant for a CoC are responsible for the quality assurance requirements as they apply to the design, fabrication, and testing of a spent fuel storage cask." However, the regulation also clearly states that, "the licensee and the certificate holder are also simultaneously responsible for these quality assurance requirements through the oversight of contractors and subcontractors." The regulation also recognizes that there are circumstances when licensees delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the quality assurance program, but that the licensee shall retain responsibility for the program. The problems and deficiencies discussed above could have been avoided had each applicant/licensee ensured that both contractor and subcontractor organizations understood and effectively implemented a QA program in accordance with 10 CFR Part 72. Oversight by an applicant/licensee should be based on direct knowledge of specific implementation of a QA program, independent evaluation of program findings, and trending of defects and problems.

If a design or fabrication problem occurs with a spent fuel storage cask the ultimate responsibility for assuring the quality of the cask still remains with the cask user, the licensee.

This information notice requires no specific action nor written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate regional office.

/RA/M. Wayne Hodges
For

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

/RA/

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

Technical Contacts: Chet Poslusny, NMSS
301-415-1341
E-mail: cxp1@nrc.gov

Charles Petrone, NRR
301-415-1027
E-mail: cdp@nrc.gov

INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/08/07	

EVENT TITLE

RESOLUTION OF GENERIC SAFETY ISSUE B-55, “IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES” (NRC Regulatory Issue Summary 2000-12)

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 United States Nuclear Regulatory Commission staff’s resolution of Generic Safety Issue (GSI) B-55, “Improved Reliability of Target Rock Safety Relief Valves." Target Rock safety relief valves (SRVs) are currently installed in the main steam systems of 22 boiling-water reactors (BWRs). There have been several occurrences of improper operation of these SRVs both in spuriously opening and blowing down the reactor coolant and in opening at pressures significantly above technical specification requirements. As a result of the actions that have been taken by BWR Owners Group and the individual BWR licensees to improve the performance of Target Rock SRVs, the staff has closed GSI B-55. The issue closeout is documented in a memorandum from S.J. Collins, Director, NRC Office of Nuclear Reactor Regulation to W.D. Travers, NRC Executive Director for Operations, dated December 17, 1999, which may be found in the NRC Agencywide Documents Access and Management System (ADAMS) under Accession Number ML993620214. In summary, the staff determined that the BWR Owners Group and the licensees have significantly improved the performance of the three-stage and two-stage Target Rock SRVs and that they are continuing to evaluate and improve the performance of the SRVs, as necessary, with sufficient resources. Therefore, the staff found that no new requirements were necessary as a result of this generic issue.

RESOLUTION OF GENERIC SAFETY ISSUE B-55, “IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES” (NRC Regulatory Issue Summary 2000-12)

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, DC 20555-0001

August 7, 2000

NRC REGULATORY ISSUE SUMMARY 2000-12
RESOLUTION OF GENERIC SAFETY ISSUE B-55, "IMPROVED RELIABILITY OF TARGET ROCK SAFETY RELIEF VALVES"

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 addressees about the staff's resolution of Generic Safety Issue (GSI) B-55, "Improved Reliability of Target Rock Safety Relief Valves." This RIS does not transmit any new requirements or staff positions. No specific action or written response is required.

BACKGROUND INFORMATION

Target Rock safety relief valves (SRVs) are currently installed in the main steam systems of 22 boiling-water reactors (BWRs). There have been several occurrences of improper operation of these SRVs both in spuriously opening and blowing down the reactor coolant and in opening at pressures significantly above technical specification requirements. There are two different designs of Target Rock SRVs. The earlier design is the three-stage SRV, which has had a history of spuriously opening and failing to reseat. This behavior was exhibited during several events which mostly occurred in the 1970s. The later design is the two-stage SRV, which is a modification of the three-stage SRV and was designed to eliminate the spurious opening and blowing down problem. Currently, there are 11 BWRs that use three-stage SRVs, and 11 BWRs that use two-stage SRVs.

GSI B-55 was prioritized by the NRC staff as a "medium" priority issue on the basis of concerns about the three-stage SRVs. Beginning in 1978, two-stage SRVs were installed in several BWRs and during operation and surveillance testing, these two-stage SRVs had problems with opening at pressures exceeding the technical specification limits. As a result, the staff also included this upward setpoint drift problem in GSI B-55 for resolution.

ML003726865

SUMMARY OF ISSUE

The BWR Owners Group (BWROG), General Electric Company (GE), and the individual licensees undertook several actions to improve the performance of three-stage and two-stage SRVs. For the three-stage SRVs, licensees began implementing GE Service Information Letter (SIL) 196, Supplement 3, which recommended increasing the SRV setpoints to raise the simmer margin between the normal reactor pressure and the SRV setpoints. Licensees also began implementing more frequent maintenance and testing of the three-stage valves. These activities were aimed at reducing the three-stage pilot valve seat leakage and have been shown to be effective through successful operating experience over many years. For the two-stage SRVs, the primary cause of the upward setpoint drift problem was determined to be corrosion bonding of the pilot valve disk to its seat. To reduce or counteract the corrosion bonding, several design improvements were developed and implemented, and three different modifications were ultimately found to significantly improve performance. These three modifications are (1) the installation of ion beam implanted platinum pilot valve disks, (2) the installation of Stellite 21 pilot valve disks, and (3) the installation of additional pressure actuation switches. The ion beam implanted platinum and Stellite 21 pilot valve disks have improved the performance of the two-stage SRVs by reducing the corrosion bonding of the pilot valve disks to their seats. Additional pressure actuation switches counteract the effects of corrosion bonding by actuating the SRVs at the proper setpoints with external air power.

As a result of the actions that have been taken by BWROG and the individual BWR licensees to improve the performance of Target Rock SRVs, the staff has closed GSI B-55. The issue closeout is documented in a memorandum from S.J. Collins, Director, NRC Office of Nuclear Reactor Regulation to W.D. Travers, NRC Executive Director for Operations, dated December 17, 1999, which may be found in the NRC Agencywide Documents Access and Management System (ADAMS) under Accession Number ML993620214. In summary, the staff determined that the BWROG and the licensees have significantly improved the performance of the three-stage and two-stage Target Rock SRVs and that they are continuing to evaluate and improve the performance of the SRVs, as necessary, with sufficient resources. Therefore, the staff found that no new requirements were necessary as a result of this generic issue. If, in the future, the staff finds that actions need to be taken to improve the performance of these SRVs, the existing quality assurance, maintenance rule, and codes and standards regulations (i.e., 10 CFR Part 50, Appendix B; 10 CFR 50.65; and 10 CFR 50.55a) provide the staff with regulatory mechanisms for pursuing additional improvements, if needed, on a plant-specific basis.

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, and the public was afforded numerous opportunities to comment in meetings with the BWR Owners Group as the matter was being studied, and at the NRC staff's presentation to the Advisory Committee on Reactor Safeguards on October 1, 1999, on the proposed resolution of GSI B-55.

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

/RA/Charles E. Ader FOR

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

Technical contact: Gary Hammer, NRR
301-415-2791
E-mail: cgh@nrc.gov