

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

February 1, 1994

NRC INFORMATION NOTICE 94-08: POTENTIAL FOR SURVEILLANCE TESTING TO FAIL TO DETECT AN INOPERABLE MAIN STEAM ISOLATION VALVE

Addressees

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

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice (IN) to alert addressees to a potential for surveillance testing to fail to detect that a main steam isolation valve is mechanically bound and will not close. 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 April 19, 1993, while performing maintenance to repair a presumed faulty limit switch on a main steam isolation valve, the licensee for the River Bend Station (River Bend) found that the valve was mechanically bound and would not close as required. The valve is a 24-inch-diameter (nominal), spring and pneumatic closing, pneumatic-opening, internally balanced, poppet-type globe valve manufactured by the Atwood & Morrill Company Inc. Plant operators had previously performed partial stroke surveillance testing of the valve on February 27 and April 1, 1993, but did not detect that the valve would not close. The licensee later determined that the testing failed to detect that the valve was inoperable because the test did not adequately consider the design of the valve and the positioning of the limit switch arm in relation to the valve poppet travel.

The licensee determined that the valve would not close because improper clearances between the valve poppet and the valve body had caused excessive wear of the guide ribs and resulted in the valve poppet becoming mechanically bound. The excessive wear may have been avoided had the licensee installed an anti-rotation modification recommended by the manufacturer in 1989. Subsequent to this event, the manufacturer reported the failure to close to the NRC under Part 21 to Title 10 of the Code of Federal Regulations and informed affected licensees of the failure mechanism and recommended actions to prevent recurrence. NRC Inspection Report 50-458/93-18 and Licensee Event Report 93-006 provide additional details on the valve failure.

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updated on 2/4/94

PDR I&E

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## Discussion

The original design positioning of the limit switches was such that, during partial stroke testing, the limit switches could be actuated and indicate movement of the main valve poppet even though the valve poppet had not actually moved. Thus, a failure of the valve to properly stroke may go undetected by partial stroke testing.

Under normal operation, the valve stem travels 28 centimeters [11 inches] to fully stroke in either direction. As the valve strokes open, the first 2.5 centimeters [1 inch] of stem travel moves an internal poppet which opens an equalizing port allowing the pressure on both sides of the main poppet to equalize. During the remainder of the open stroke, the internal poppet lifts the main poppet and retracts it to the fully open position. During a partial stroke test in the close direction, as the stem (and the internal poppet) begins to close, the main poppet also begins to close because of gravity. However, during the event, with the main poppet stuck in the open position, the stem travelled about 2.5 centimeters [1 inch] and stopped when the internal poppet seated in the equalizing port.

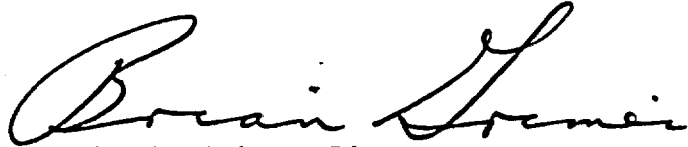
There are three limit switches on the valve that are of concern in this event. The first two switches provide a safety-related signal to the reactor protection system that the valve is 92 percent open. The third switch sends a nonsafety-related signal to position indicating lights in the control room indicating that the valve is 90 percent open. The licensee had set the 90-percent-open limit switch such that stem movement of about 2.8 centimeters [1.1 inch] was required to actuate the switch and indicate that the main valve poppet had moved to the 90-percent-open position. However, because the limit switches are set with a  $\pm 2$  percent tolerance, actual stem travel to actuate the 90-percent-open limit switch may be only 2.25 centimeters [0.88 inch]. In a worst-case scenario both the 92- and the 90-percent-open limit switches could be actuated without the main valve poppet moving.

During the partial stroke testing conducted on February 7 and April 1, the first two limit switches (92-percent-open indication) actuated, the third limit switch (90-percent-open indication) did not actuate. Although the procedural step called for receipt of the 90-percent-open indication, the operators did not declare the valve inoperable because the first two limit switches had actuated and they assumed that the third limit switch (nonsafety) had failed. Later, on April 17, during maintenance on the presumed faulty limit switch, the licensee found that the main valve poppet was mechanically bound and that the valve would not close.

The licensee for River Bend changed the third limit switch setting to actuate at 85 percent of the open position to ensure that its actuation during partial stroke testing would give positive indication of poppet movement. Pending further evaluation of these valves during the next refueling outage, the licensee is performing full stroke testing of the valves on a quarterly basis and intends to install the anti-rotation modification recommended by the vendor to prevent recurrence of the excessive wear of the valve guides. Also, operations personnel have been trained on the operation and function of the limit switches.

The valve described in this notice is used in safety-related applications at nuclear facilities. One such application is as a main steam isolation valve. At River Bend and most domestic boiling water reactors, there are two main steam isolation valves for each main steam line; an inboard valve, located inside the drywell, and an outboard valve, located just outside the primary containment. After a design-basis accident, these valves are required to close and remain closed for 1 hour. Should these valves fail to close, offsite dose limits could be exceeded. A similar failure of a main steam isolation valve to close had occurred at a foreign boiling water reactor. The potential for limit switch positioning to adversely affect surveillance test accuracy may exist for valves other than that described in this notice.

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



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

Technical contacts: Thomas F. Westerman, RIV  
(817) 860-8145

Patricia Campbell, NRR  
(301) 504-1311

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(512) 972-2507

William M. McNeill, RIV  
(817) 860-8174

Attachment:  
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**\*SEE PREVIOUS CONCURRENCE**

OFFICE	RPB:ADM	EMEB:DE:NRR	C/EMEB:DE:NRR	PDIV-2
NAME	RSanders*	PCampbell*	JNorberg*	EBaker*
DATE	11/18/93	12/21/93	12/22/93	12/27/93
REGION IV	REGION IV	C/ES:RIV	DD/DRS:RIV	D/DRS:RIV
WMcNeill*	DLoveless*	TWesterman*	AHowell*	SCollins*
12/27/93	01/03/94	01/03/94	01/03/94	01/04/94
OIP/NA/Per	OGCB:DORS:NRR	C/OGCB:DORS:NRR	D/DORS:NRR	
KHenderson*	JBirmingham*	GHEMarcus*	BGrimes	
01/06/94	01/10/94	01/10/94	01/27/94	

OFFICIAL DOCUMENT NAME: 94-08.IN

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OFFICIAL DOCUMENT NAME: ATWOODIN.JLB

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 NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-26, Supp. 1	Grease Solidification Causes Molded-Case Circuit Breaker Failure to Close	01/31/94	All holders of OLs or CPs for nuclear power reactors.
94-07	Solubility Criteria for Liquid Effluent Releases to Sanitary Sewerage Under the Revised 10 CFR Part 20	01/28/94	All byproduct material and fuel cycle licensees with the exception of licensees authorized solely for sealed sources.
94-06	Potential Failure of Long-Term Emergency Nitrogen Supply for the Automatic Depressurization System Valves	01/28/94	All holders of OLs or CPs for boiling water reactors.
93-85, Rev. 1	Problems with X-Relays in DB- and DHP-Type Circuit Breakers Manufactured by Westinghouse	01/20/94	All holders of OLs or CPs for nuclear power reactors.
94-05	Potential Failure of Steam Generator Tubes with Kinetically Welded Sleeves	01/19/94	All holders of OLs or CPs for pressurized water reactors (PWRs).
94-04	Digital Integrated Circuit Sockets with Intermittent Contact	01/14/94	All NRC licensees except licensed operators.
94-03	Deficiencies Identified during Service Water System Operational Performance Inspections	01/11/94	All holders of OLs or CPs for nuclear power reactors.
94-02	Inoperability of General Electric Magne-Blast Breaker Because of Misalignment of Close-Latch Spring	01/07/94	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License  
 CP = Construction Permit

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