TELEPHONE (815) 727-2600
POST OFFICE BOX 3338
JOLIET, IL 60434

CRANE NUCLEAR OPERATIONS

CRANE VALVES • 104 NORTH CHICAGO STREET • JOLIET, IL 60431

August 30, 1994

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

SUBJECT: *POTENTIAL 10CFR21 CONDITION*

REFERENCE: *a.) Crane Letter dated August 23, 1994*
b.) Northern States Power Letter to the NRC dated April 18, 1994

Gentlemen:

This letter will serve to advise you of our continuing investigation concerning Crane design Gate Valves involved in the January, 1994 incident at the Prairie Island Nuclear Generating Plant. Details of that incident were contained in the reference (b) correspondence, a copy of which as attached.

We have been in constant communication with Westinghouse Electric's NATD Engineering Technology Department concerning their December, 1969, P.O. 546-CAK-116878 BN on which the valves in question were purchased from Crane Company. In turn, Westinghouse supplied the valves to a number of Nuclear Utilities, records for which have been retrieved from Crane archives.

On August 23, 1994, a letter was issued by Crane to those Utilities to whom valves were supplied. A sample copy of the letter is attached for your reference and a total list of Utilities to whom the letter was sent is shown on the attached page.

At this point, we consider the incident a potentially reportable condition based on the NSP evaluation at their plant. Upon receipt of additional input from the balance of the plants, we will assess the situation to determine a subsequent course of action. Historical records do not contain information addressing the total number of valves of this configuration manufactured by Crane, nor their application throughout the Industry.

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As additional information becomes available, we will advise. You may direct any questions to the writer or to our Manager of Developmental Engineering, Mr. Bruce Harry. My direct line is 815-740-7597 and Bruce can be reached at 815-740-7570.

Sincerely,

R. F. Hornyak

Ronald F. Hornyak
Manager QA/Support Engineering

RFH/cs

cc: *J. Carlson*
F. Bisesto
B. Harry
K. Hutchinson
H. Sandner (Westinghouse)
File

Attachments

TELEPHONE (815) 727-9800
POST OFFICE BOX 3379
JOLIET, IL 60434

CRANE NUCLEAR OPERATIONS

CRANE VALVES • 104 NORTH CHICAGO STREET • JOLIET, IL 60431

August 23, 1994

Beaver Valley Power Station
SEB-3
Duquesne Light Company
P.O. Box 4
Shippingport, PA 15077

Gentlemen:

In January of this year a containment isolation valve at the Prairie Island Nuclear Station was damaged as a result of operational problems involving maintenance procedures. Details of the event have been reported to the NRC via Prairie Island's Docket Nos. 50-282 and 50-306 and associated NRC Inspection Report Nos. 282/94002 and 306/94002.

The valve in question was a Crane 10" Figure 63174 and during the course of the investigation, it was discovered that, contrary to information shown on the valve assembly drawing, shims were installed between the yoke and the adapter plate. The extraordinary conditions experienced by the valve during the event resulted in failure of the welds joining the yoke and adapter plate and more catastrophic failure to the other valve parts.

Crane assisted the Utility in evaluating additional valves of the same design for evidence of the existence of shims and for the adequacy of the attachment welds where shims did exist. It was concluded that a substantial safety hazard did not exist at the Prairie Island plant, based on the walk-down that was performed and additional analytical evaluations of the as-built valve characteristics.

Since that incident, Crane has conducted additional evaluations, including a search of archives stored in a Crane facility in Pennsylvania. The valves at the Prairie Island plant were part of an order produced by Crane for Westinghouse Electric in the early 1970's. Although the records retrieved contained no information to support the addition of shims to the design, the records indicate that valves of the Figure 63174 design were supplied to you on that same Westinghouse order.

Crane requests your cooperation in determining if those valves are in your system. We will be glad to assist you in evaluating the condition of the hardware and in performing any analytical evaluations to determine if safety hazards exist. Valves of the 63174 design were supplied in 10" and 12" sizes.

You have been contacted based upon your affiliation with the Motor Operated Valve User's Group and its Member Roster. We appreciate your cooperation in channeling this communication to the appropriate personnel at your facility.

Your questions can be directed to the writer or to our Manager of Development Engineering, Mr. Bruce Harry (who performed the initial on-site evaluation at Prairie Island) or Mr. David Dwyer, our Analytical Project Engineer, who has paid subsequent visits to the Prairie Island plant. Our general number in Joliet is 815-727-2600.

Very Truly Yours,

R.F. Hornyak

R.F. Hornyak

Manager QA/Support Engineering

RFH/lk

cc: *J. Carlson*
K. Hutchinson
B. Harry
D. Dwyer

Attachment

Names / Address of Recipients of August 23, 1994 Letter

✓ *Beaver Valley Power Station*
SEB-3
Duquesne Light Company
PO Box 4
Shippingport, PA 15077

ATTN: Mr. Neil Morrison
Senior Engineer

✓ *Diablo Canyon Power Plant*
Pacific Gas & Electric Co.
PO Box 56
Avila Beach, CA 93424

ATTN: Mr. Don Bauer

✓ *Corporate Office*
BR 5A
Tennessee Valley Authority
1101 Market St.
Chattanooga, TN 37402-2801

ATTN: Mr. Richard G. Sim[illegible]
Program Manager, Va[illegible]

✓ *Innsbrook Technical Center*
Virginia Power Company
5000 Dominion Blvd.
Glen Allen, VA 23060

ATTN: Ms. Pamela E. Detine
Project Engineer

✓ *Kewaunee Nuclear Power Plant*
Wisconsin Public Service Corp.
N 490 Hwy 42
Kewaunee, WI 54216

ATTN: Mr. Larry L. Limberg
Maintenance Engineer

MC N-50
✓ *Public Service Electric & Gas Company*
PO Box 236
Hancocks Bridge, NJ 08038

ATTN: Mr. Robert S. Lewis
Senior Staff Engineer

Northern States Power Company

Prairie Island Nuclear Generating Plant

1717 Wakonade Dr. East
Welch, Minnesota 55089

April 18, 1994

10 CFR Part 2
Appendix C

US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Reply to a Notice of Violation
NRC Inspection Report Nos. 282/94002(DRP) and 306/94002(DRP)
Procedural Deficiency Allowing Damage to a Safety-related Valve

Your letter of March 18, 1994 transmitted the subject inspection report and violation notice which required a 30 day response. Attached is our response.

In our response, we have made new NRC commitments which are identified as such in the attachment as the statements which are in italics.

If you have any questions regarding this response, please contact Jack Leveille (612-388-1121, extension 4562).

E. L. Watt
General Manager
Prairie Island Site

cc: Regional Administrator III, NRC
Senior Resident Inspector, NRC
NRR Project Manager, NRC
J.E. Silberg

Attachment: RESPONSE TO VIOLATION

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RESPONSE TO VIOLATION

Violation

Criterion V of 10 CFR 50, Appendix B, requires that activities affecting quality be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on January 24, 1994, extensive damage to the normally closed outboard containment isolation valve (MV-32181) between the containment building sump and the suction of No. 22 residual heat removal pump occurred when the electrician, stroking the valve locally from the motor control center, continually depressed the close contactor instead of depressing the open contactor. The maintenance procedure used to locally stroke the valve was not appropriate to the circumstances in that: 1) the procedure did not require that direct communications be established between the electrician and operators during stroking of the valve; 2) expected values for motor current draw were not included; 3) the method of making up the contactor was not specified (i.e. the open contactor did not have to be continually depressed in order to operate the valve); and 4) the procedure required errorless human performance because of the absence of actuator protective features.

This is a Severity Level IV Violation (Supplement I).

Background

We consider this failure to be significant since a risk-significant component failed, albeit not while performing in the accident functional mode.

Although the work procedure for cycling the valve had the deficiencies noted in the violation, it should be noted that the plant electricians have cycled motor valves locally at the breaker using the contactors numerous times as part of motor-operated valve (MOV) testing efforts. Such cycling activities have occurred since the 1980's. In an effort to reduce the likelihood of error, the motor valve and system engineers initiated the procedure referenced in the violation. In preparing the procedure the engineer had spoken with electrical maintenance personnel [illegible]erienced in motor operated valve testing to ensure that the procedure would be appropriate. Based on this input, the procedure was developed for cycling MV-32181.

MV-32181 was the third valve to be cycled in response to a NRC identified concern regarding the potential for pressure locking. The person assigned to perform the task was a journeyman electrician. Previously in January, two different electricians had used the same procedure satisfactorily on the two Unit 1 valves.

Reason for the Violation

Two root cause analyses were initiated - one to determine the root cause for the event and the second to determine the valve failure root cause (this is discussed in the "Corrective Steps Taken and Results Achieved" section). The event root cause analysis was performed by the on-site Error Reduction Task Force (ERTF) and is documented in ERTF Report 94-01.

The ERTF report identified two primary causes, two secondary causes, and two additional possible causes for the event. The primary cause was determined to be human error. The causes were:

Primary causes - Human Error

1. Self-checking was not applied to verify that the choice of contactor was correct, or that the intended action was correct. The electrician depressed the wrong contactor.

2. The electrician did not have the proper information at the job site to verify whether the valve's circuit was seal-in or not. Without this information, the electrician pressed and held the contactor to ensure the valve would go open for the required 30 seconds. Holding the contactor in bypasses the torque switch trip.

Secondary causes - Ergonomic

1. The open and close contactors in the MCC breaker cubicle were not labeled.

2. The work request was somewhat generic in that it did not specify the expected current draw. Also, the work request did not contain instructions to push and release the contactor, nor did it mention the seal-in feature.

Possible contributing causes

1. No communications were established between the MCC breaker cubicle and the motor valve.

2. Consequences of potential error were not discussed before starting the work.

Corrective Steps Taken and Results Achieved

The affected valve was repaired and restored to service on January 26, 1994. Prior to cycling additional motor operated valves (e.g., MV-32180, Containment

Sump B to 21 EHR Pump) by this method, the causes of the event were identified and corrective actions to prevent a similar event were discussed with the plant electricians. A more detailed procedure was used f' the cycling of the next valve, MV-32180, which was cycled successfully on February 4, 1994.

Prairie Island has developed a videotape intended to emphasize self-checking. This video, "Right from the Start", has now been viewed by some, but not all, of the plant staff, including the electricians.

A method was implemented to label the open and close contactors in motor valve MCC breaker cubicles. To date, the contactors in 142 MCC breaker cubicles have been labeled.

On March 24, 1994, temporary memos (94-24 and 94-25) were issued to both units' quarterly surveillance procedures, SP1089 and SP2089 (Residual Heat Removal Pumps and Suction Valves from the Refueling Water Storage Tank), respectively. These temporary memos are refinements of the procedure used for the MV-32180 cycling. These refinements were developed during a post-event evaluation by those involved in the event. These procedure changes involved cycling the Sump B valves locally for potential pressure locking. The procedure specified the following additional information/requirements beyond those specified in the original work request:

(1) precaution to self check,

(2) perform a pre-job briefing,

(3) use of headset communication between the electrician at the MCC breaker cubicle and the operator at the valve,

(4) identification of the expected full load and nameplate locked rotor amperage,

(5) caution that the contactor need not be held in, but only momentarily depressed since it is a seal-in circuit, and

(6) verification that the open and closed contactors are labeled.

The MOV testing engineers were advised to ensure adequate instructions are provided for those cases where local operation of an MOV is required. They will evaluate their procedures for necessary changes.

A Safety Evaluation revision provides short term justification of the operability of the Sump B valves based on the calculational methodology of required opening thrusts under pressure locking condition. These valves will no longer be cycled for pressure locking concerns. A modification is being prepared to modify the valves to prevent pressure locking.

The Equipment Failure Root Cause evaluation was initiated by the engineering

staff with the assistance of the valve manufacturer (Crane/Aloyco), an independent engineering firm (Altran, experienced in failure analysis of this type, and the corporate Materials and Special Processes department. The vendor analyses are still in progress.

Mechanically the weld failed at the point where the yoke arm was attached to the actuator adapter plate. This failure location was not predicted by the valve manufacturer's weak link analysis. Rather, failure was anticipated in the yoke. The failure root cause analysis has determined that the following additional factors contributed to the premature failure:

(1) A shim was installed between the yoke and adapter plate that resulted in a lower failure stress than design.

(2) The shim was not indicated on the fabrication drawings.

(3) The shim was not included in the design calculations, and

(4) The weld that broke was poor quality as indicated by less-than-design fusion.

Preliminary data from Crane and Altran indicate that the as-built values for the weldments were less than the original Westinghouse specification design values. These additional factors are apparently due to inadequate Crane quality assurance and controls and inadequate Westinghouse oversight. Crane was the manufacturer and Westinghouse was the supplier.

We have determined that the as-built valve characteristics do not constitute a substantial safety hazard for the Prairie Island plant application. However, the existence of the shims in valves in different applications in other plants may present a substantial safety hazard. We have notified Westinghouse that they may need to perform an evaluation for 10 CFR Part 21 reporting purposes.

Although the torque switch was bypassed due to the actions of the electrician, the torque switch settings could have been set at a lower setting since the plant design differential pressure is 46 psid rather than the generic 700 psid design differential pressure.

The discovery of the shim led to expansion of the investigation to all other Crane valves of this type used in safety-related applications. These valves include:

Containment Sump B to RHR:	12"	32075, 32076, 32077, 32078	(Unit 1)
		32178, 32179, 32180, 32181	(Unit 2)
RWST to RHR Pumps:	10"	32084, 32085	(Unit 1)
		32187, 32188	(Unit 2)

The four RWST to RHR Pump valves and the other three outside Sump A to RHR

valves were inspected for presence of a shim and weld quality. The results of the inspections were satisfactory. The four inside Sump B to RHR valves were not inspected at this time since a unit shutdown would be needed and it is believed that inspection of the remaining valves provides a reasonable expectation that the valves inside the containment boundary are not different.

The torque switch setting for MV-32181 has been adjusted downward to correlate closer to the 46 psid plant design than the higher generic design differential pressure. This change had been planned prior to the event occurrence.

A further investigation of the maintenance history showed that one of the Unit 1 RWST to RHR pump valves failed in 1975. At that time those four valves were evaluated and reveldad as appropriate.

Corrective Steps That Will Be Taken To Avoid Further Violations

The need for a revision of the plant maintenance procedure writers' guide will be reevaluated, by June 1, 1994, in light of the observations made during the evaluation of this event.

A Maintenance section procedure will be completed, by June 1, 1994, that describes in detail the method for cycling a motor valve locally by using open and close contactors since this approach is used in the MOV testing program.

The remaining containment sump B valves will be inspected for presence of the shim and weld quality and the torque switch settings will be readjusted to correlate with the lower plant design differential pressure, by July 1, 1994 for Unit 1 and July 1, 1995 for Unit 2.

The videotape "Right from the Start" has been incorporated into the General Employee Training re-qualification program, which is presented to all personnel badged for access to the plant.

Date When Full Compliance Will Be Achieved

Full compliance has been achieved.