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IN 85-59

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

July 17, 1985

IE INFORMATION NOTICE NO. 85-59: VALVE STEM CORROSION FAILURES

Addressees:

All nuclear power reactor facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

This information notice is provided to alert recipients of a potentially significant problem pertaining to stress corrosion failures of valve stems and shafts; items that are not routinely examined. It is suggested that recipients review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

There have been four instances where cracks were found in 410 stainless steel valve stems. These instances involved different licensees and different manufacturers. Such cracks cannot be observed without the disassembly of the valves, and the valve operability test programs do not provide a means of early detection. In three of these instances, the cracks grew until the stem sheared when the valve was activated. Such failures can prevent the system from performing its safety function.

Uncontrollable leakage from the stem packing of several Velan globe valves was reported by Oconee 1 in December 1971. Disassembly and examination revealed cracks for the entire length of the stems and more than half of the diameter in depth. In order to prevent cracking, 410 stainless steel needs to be tempered immediately after hardening, but several batches were not tempered. Ultimately, 2600 stems were replaced in the 1-1/2-, 1- and 1/2-inch valves, using 17-4PH and 300-series stainless steel materials.

A 20 inch Anchor/Darling gate valve stem snapped while being manually opened at Brunswick 2 on August 4, 1982. There was pitting of the 410 stainless steel in the gland packing section and the crack had initiated from one of these pits. The cross-section area of the stem of the suppression pool suction valve had been reduced by 70% by intergranular stress corrosion cracking (IGSCC). The material had a higher hardness than specified as a result of improper heat treatment.

Excessive hardness is associated with cracking and corrosion. The manufacturer replaced five lots of valve stems with properly heat-treated 410 stainless steel, and there have not been any further problems.

An injection valve in the low-pressure coolant injection (LPCI) system broke in two places during disassembly at Browns Ferry 3 on February 28, 1984. One break was below the stem packing area and the other was at the gate connection. Over 50% of the cross-section of the stem of these 24-inch Walworth valves had been lost in these areas by IGSCC. The stem had higher hardness than specified. New stems made from 17-4 PH stainless steel were installed.

Linear indications were discovered on three main steam isolation valve (MSIV) shafts at Farley 1 on February 29, 1984. The indications were from 1 to 13 inches long, contained thick oxides, and were located in the packing gland area. The MSIVs were Atwood-Morrill 32-inch swing check valves and the shaft hardness exceeded specifications. 17-4 PH stainless steel also was used as the replacement material for the shafts.

Although 410 stainless steel is defined as a stainless steel because of its alloy content, it is really a high chromium, very hardenable steel. Cooling this material in air from the 1700-to-1900°F temperature range results in a surface hardness of up to HRC 45 and high internal stresses. Tempering the hard and brittle martensite produces a softer and more ductile composition that has much less chromium available for intergranular corrosion resistance. Tempering in the 700-to-1050°F range is not recommended because it results in low and erratic impact properties and poor resistance to corrosion and stress corrosion.

The following conclusions were reached:

1. The actual hardness of the 410 stainless steel valve stems and shafts was higher than specified and higher than documented.

2. The excessive hardness is associated with intergranular stress corrosion cracking.

3. The cracking occurred in internal areas where there could be concentrations of corroding chemicals, such as at the gland packing.

4. The oxides found in the cracks showed that the cracks occurred during service and grew slowly.

5. The cracks were not detected by the routine valve operability test programs, but were only discovered by actual failures or after disassembly during refueling outages.

6. Failure of these valves would make the specific safety system inoperable.

No specific action or written response is required by this information notice. If you have any questions about this matter, please contact the Regional Administrator of the appropriate regional office or this office.

E. L. Jordan

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

Technical Contact: P. Cortland, IE
(301) 492-4175

Attachment: List of Recently Issued IE Information Notices

Attachment 1
IN 85-59
July 17, 1985

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
85-58	Failure Of A General Electric Type AK-2-25 Reactor Trip Breaker	7/17/85	All power reactor facilities designed by B&W and CE holding an OL or CP
85-57	Lost Iridium-192 Source Resulting In The Death Of Eight Persons In Morocco	7/16/85	All power reactor facilities holding an OL or CP; fuel facilities; and material licensees
85-56	Inadequate Environment Control For Components And Systems In Extended Storage Or Layup	7/15/85	All power reactor facilities holding an OL or CP
85-55	Revised Emergency Exercise Frequency Rule	7/15/85	All power reactor facilities holding an OL or CP
85-54	Teletheraphy Unit Malfunction	7/15/85	All NRC licensees authorized to use teletheraphy units
85-53	Performance Of NRC-Licensed Individuals While On Duty	7/12/85	All power reactor facilities holding an OL or CP
85-52	Errors In Dose Assessment Computer Codes And Reporting Requirements Under 10 CFR Part 21	7/10/85	All power reactor facilities holding an OL or CP
85-51	Inadvertent Loss Or Improper Actuation Of Safety-Related Equipment	7/10/85	All power reactor facilities holding an OL or CP
85-50	Complete Loss Of Main And Auxiliary Feedwater At A PWR Designed By Babcock & Wilcox	7/8/85	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit