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

January 4, 1993

NRC INFORMATION NOTICE 93-02: MALFUNCTION OF A PRESSURIZER CODE SAFETY 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 to alert addressees to the failure of a pressurizer code safety valve to maintain set pressure and reseat properly during a plant transient. 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 July 3, 1992, after an electrical transient, the reactor at Fort Calhoun Station tripped on high primary pressure. Both power-operated relief valves opened and valve RC-142, one of two pressurizer code safety valves, lifted prematurely at a pressure below 16.75 MPa [2430 psia], as opposed to the proper setpoint pressure of 17.24 Mpa [2500 psia] +/- 1 percent. The relief valves shut automatically when the reactor coolant system pressure decreased to 16.20 Mpa [2350 psia]. Because a safety valve was still open, the pressure continued to decrease and RC-142 subsequently reseated at approximately 12.03 MPa [1745 psia]. The pressure then increased and RC-142 lifted again at approximately 13.27 MPa [1925 psia]. RC-142 partially reseated, as pressure again dropped, at approximately 6.89 MPa [1000 psia]. RC-142 continued to leak, as indicated by the tail pipe temperature, until the plant was brought to cold shutdown. The licensee removed RC-142 and sent it to Wyle Laboratory (Wyle) for inspection and testing.

Discussion

The code safety valves installed at Fort Calhoun are "3-inch inlet by 6-inch outlet", Size 3K6, Style HB-86-BP, Type E valves (Figure 1) manufactured by the Crosby Valve and Gage Company (Crosby). In early 1980, the Electric Power and Research Institute (EPRI) tested Crosby safety valves that have loop seals and are subjected to back pressure. The EPRI test results indicated that the initial discharge of the loop seal or a transition from discharging steam to discharging water could cause the valve to chatter.

The licensee believes that RC-142 chattered during its initial lift from the discharge of the loop seal. Apparently, the chatter loosened the locknut on the adjusting bolt and allowed the adjusting bolt to partially back out. Later, primary water discharged through RC-142 for approximately 5 minutes (pressurizer level reached 100 percent) during its second lift and subsequent partial reseat at approximately 6.9 MPa [1000 psia]. The discharging water induced further chattering, apparently causing the adjusting bolt to back out even further, reducing the valve lift setpoint to approximately 10.18 MPa [1477 psia].

To ensure that the adjusting bolt would not back out again, Crosby designed a special mechanical locking device and installed it on the two valves at Fort Calhoun. Crosby also specified a torque value of "400 foot-pounds" for the adjusting bolt locknut. This value had not been previously specified in procedures used by Wyle for inspecting and testing pressurizer code safety valves.

An NRC augmented inspection team monitored licensee activities at Fort Calhoun and Wyle Laboratory. At Wyle, the locknut for RC-142 was found to have backed off from the top of the valve bonnet by approximately 3 to 6 mm [1/8 to 1/4 inch] and could be turned by hand. The adjusting bolt was determined to be 19.5 flats of bolt revolution from the zero compression position of the spring. Crosby representatives calculated that this position corresponded to a setpoint value of approximately 10.18 MPa [1477 psia]. In March 1992, the valve had been set to 17.24 MPa [2500 psia] +/- 1 percent at Wyle Laboratory.

When the valve internals were removed, the bellows assembly was found to have failed on each end at the first weld after the transition weld. Also, the disc insert was found jammed into the disc holder. The disc insert was recessed approximately 0.05 mm [0.002 inch] below the top surface of the disc ring. The disc ring was seated on the nozzle ring, which indicated that the valve had not reseated properly. NRC Inspection Report 50-285/92-18 contains additional information on this event.

On August 22, 1992, an additional problem was revealed when RC-142 again lifted prematurely. This premature lift occurred as the reactor coolant system pressure increased to approximately 16.53 MPa [2397 psia]. This pressure was approximately 4 percent below the normal setpoint of 17.24 MPa [2500 psia] +/- 1 percent. However, the valve reseated properly with no leakage detected before or after the valve lift.

Because of the premature lift, both valves were returned to Wyle Laboratory for additional inspection and testing. Wyle inspected RC-142 and found the valve to be in good condition with only minor nicks on the nozzle seat. When Wyle attempted to test RC-142, certain conditions and difficulties were noted.

- RC-142 has a stainless steel nozzle and a carbon steel body.

- The initial attempt to test the valve under "cold" conditions to simulate the normal operating conditions of the plant could not be

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Office of Nuclear Reactor Regulation

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(817) 860-8145

P. A. Goldberg, Region IV
(817) 860-8168

Attachments:
1. Figure 1, Fort Calhoun Pressurizer Safety Valve
2. List of Recently Issued NRC Information Notices

performed as planned because of increases in the valve internal and external temperatures before and after the lift of the valve.

However, the testing revealed that the lift setpoint would increase with increasing nozzle temperature and then decrease as the valve body temperature began to increase. Licensee personnel concluded that the valves would require testing at "hot" upper bound temperatures with saturated inlet steam near setpoint pressure and with the valves insulated with the actual plant insulation. The insulation used by Wyle differed somewhat in composition and fit from the plant insulation and apparently affected the distribution of heat within the valve. The data, obtained from the tests performed with saturated steam, confirmed that the setpoint value initially increased due to the thermal expansion of the valve nozzle, then decreased as the temperature of the valve body increased. The setpoint value stabilized once the valve temperature stabilized. This condition was also observed during the "in-situ" Trevitest testing performed at the plant. As a result of this testing, the licensee performed a safety analysis and determined that the reactor coolant system could withstand an overpressure transient with a code safety valve pressure setpoint deviation of +6 percent (1.03 MPa [150 psi]) and that a pressure setpoint deviation of -4 percent (0.69 MPa [100 psi]) would not cause unnecessary challenges to the safety valves. The licensee also reduced the power-operated relief valve and reactor high pressure trip setpoint by 0.35MPa [50 psi]. The licensee is considering the removal of the loop seal as a long-term action.

After completing the testing, Wyle reset both code safety valves to the technical specification setpoint of "2500 psia +/- 1 percent." The valves were then returned to Fort Calhoun and were reinstalled.

The licensee believes that the premature lift of RC-142 on August 22, 1992, resulted from using Wyle insulation during testing rather than the actual plant insulation. The tests using Wyle insulation resulted in a lower stabilized valve temperature than resulted from using the actual plant insulation. The difference in the test methods resulted in an approximate 3 percent difference in setpoint. NRC Inspection Report 50-285/92-21 contains additional information on this event.

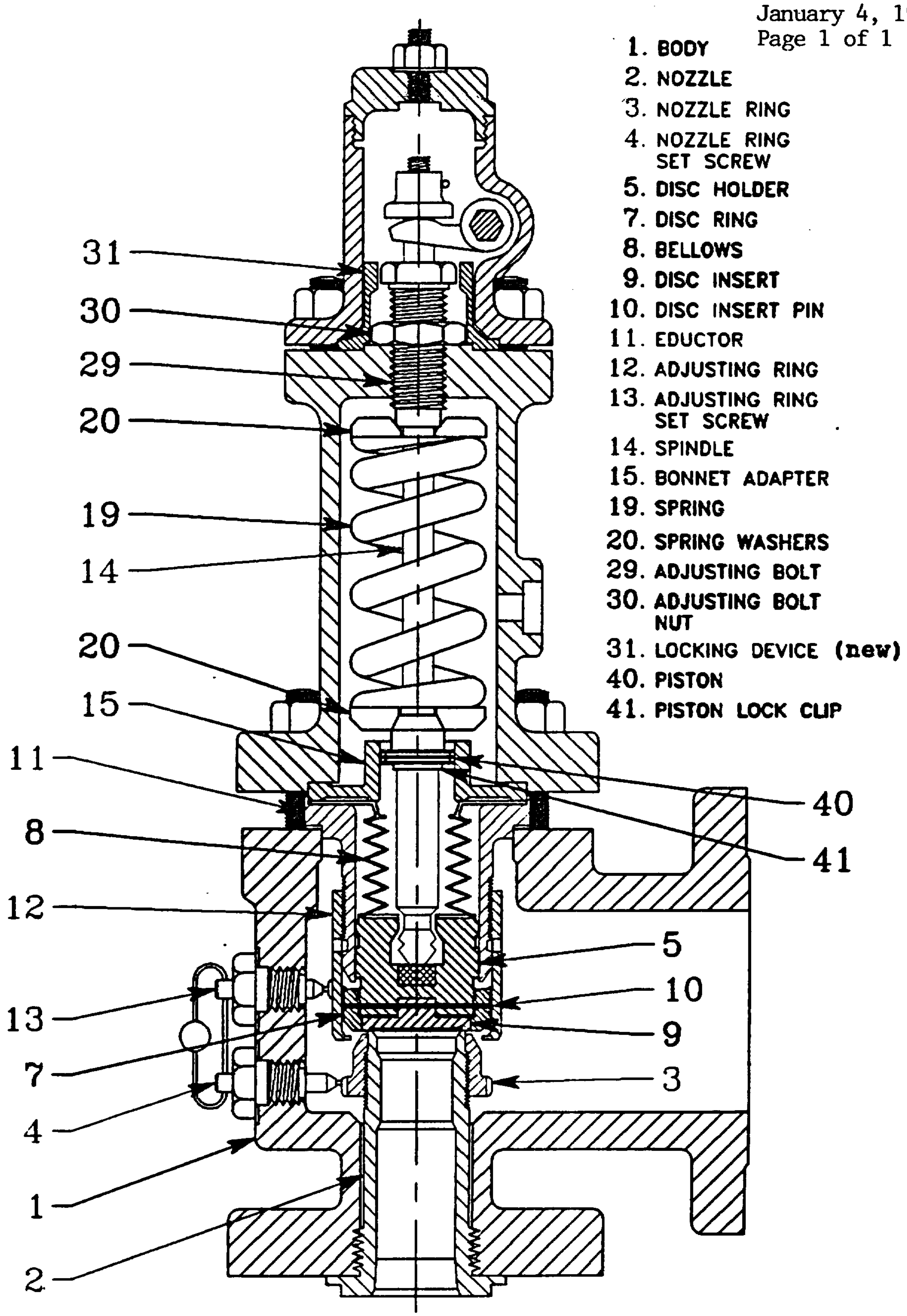

FIGURE 1

FORT CALHOUN PRESSURIZER SAFETY VALVE

LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-01	Accuracy of Motor-Operated Valve Diagnostic Equipment Manufactures by Liberty Technologies	01/04/93	All holders of OLs or CPs for nuclear power reactors.
92-86	Unexpected Restriction to Thermal Growth of Reactor Coolant Piping	12/24/92	All holders of OLs or CPs for nuclear power reactors.
92-85	Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage	12/23/92	All holders of OLs or CPs for nuclear power reactors.
92-84	Release of Patients Treated with Temporary Implants	12/17/92	All Nuclear Regulatory Commission Medical Licensees
88-23, Supp. 4	Potential for Gas Binding of High-Pressure Safety Injection Pumps during A Design Basis Accident	12/18/92	All holders of OLs or CPs for nuclear power reactors.
92-83	Thrust Limits for Limitorque Actuators and Potential Overstressing of Motor-Operated Valves	12/17/92	All holders of OLs or CPs for nuclear power reactors.
92-82	Results of Thermo-Lag 330-1 Combustibility Testing	12/15/92	All holders of OLs or CPs for nuclear power reactors.
92-81	Potential Deficiency of Electrical Cables with Bonded Hypalon Jackets	12/11/92	All holders of OLs or CPs for nuclear power reactors.
92-80	Results of Thermo-Lag 330-1 Combustibility Testing	12/07/92	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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Original signed by
Brian K. Grimes

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Attachments:
1. Figure 1, Fort Calhoun Pressurizer Safety Valve
2. List of Recently Issued NRC Information Notices

*SEE PREVIOUS CONCURRENCE

*OGCB:DORS JBirmingham:mkm 12/04/92	*OEAB:DORS JRamsey 09/29/92	*OEAB:DORS KMarcus 09/29/92	*ASC/OEAB:DORS AGautam 09/26/92	*ADM:RPB JMain 09/29/92
*RIV/DRS TWesterman 11/04/92	*RIV/DRP PHarrell 11/16/92	*C/EMEB:DE JNorberg 11/16/92	*C/OEAB:DORS AChaffee 11/23/92	
*C/OGCB:DORS GMarcus 12/15/92	D/DORS BGrimes 12/28/92			

DOCUMENT NAME: 93-02.IN

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*RIV/DRS TWesterman 11/04/92	*RIV/DRP PHarrell 11/16/92	*C/EMEB:DE JNorberg 11/16/92	*C/OEAB:DORS AChaffee 11/23/92	
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Reviewed by J. Main on 9/29/92. JER

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Attachments:
1. Crosby Pressurizer Code Safety Valve Figure
2. List of Recently Issued NRC Information Notices

OEAB:DOEA KMarcus / /92	ASG ASC/OEAB:DOEA AGautam 9/26/92	ADM:RPB JMain / /92	RIV/DRS TWesterman / /92
RIV/DRS PGoldberg / /92	RIV/DRP PHarrell / /92	EMEB:DET GHammer / /92	C/OEAB:DOEA AChaffee / /92
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