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

August 22, 1988

NRC INFORMATION NOTICE NO. 88-68: SETPOINT TESTING OF PRESSURIZER SAFETY VALVES WITH FILLED LOOP SEALS USING HYDRAULIC ASSIST DEVICES

Addressees:

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

Purpose:

This information notice is being provided to alert addressees to potentially generic problems that have occurred during testing of pressurizer safety valves with filled loop seals using a hydraulic assist device. Use of hydraulic assist devices may result in inaccurate results for safety valve setpoint testing when the valves are subjected to water or two-phase flow. 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 notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On August 30, 1986, the licensee of Diablo Canyon 1 tested the three pressurizer safety valves with the loop seal filled using a hydraulic assist device. The initial lift points were reported as 2747.8, 3028.0, and 2661.0 psig for valves number RCS-1-8010A, B, and C, respectively. The required setpoint for the valves was 2485±1% psig. The test method monitored hydraulic pressure on the test rig for an indication of valve stem displacement to infer lift point. The licensee concluded that the inferred lift point was not accurate on the first lift because the loop seal was not drained. Water moving through the seat area produced little valve stem displacement because of the different physical properties of steam and water. Steam then entered the valves at an elevated hydraulic pressure and caused a larger displacement, resulting in the prediction of an inaccurately high lift point. After the loop seal was drained, the lift points of the valves were measured again and found to be within technical specification (TS) limits (2485±1%) at 2464, 2493, and 2503 psig (LER 50/275-86/018).

On April 2, 1988, the licensee of Sequoyah 2 tested the setpoints of the pressurizer safety valves to determine if low setpoints could be the cause of the leakage that the valves had been experiencing. A hydraulic assist device

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was used with reactor pressure between 1700 psig and normal operating pressure, and with a water seal at the valve seat maintained at an elevated temperature by external heaters. The lifting force required to open the valve was measured by a load cell and recorded on a strip chart to provide data necessary to calculate the valve setpoint. Disc lift was determined by a change in slope on the load cell trace and confirmed by test personnel who listened for audible passage of flow through the discharge piping. With valve lift assumed known, the lift force was used to calculate an equivalent pressure. This pressure was added to the system static pressure to infer the valve setpoint. Initial lift results were 2634 psig for valve 2-68-563, 2678 psig for 2-68-564, and 2660 psig for 2-68-565. The required setpoint was 2485±1% psig for each valve. Setpoints were readjusted and the valves were retested in situ, after re-establishing the water seal.

On April 8, 1988, with the unit in cold shutdown, the Sequoyah 2 pressurizer safety valves were sent to Wyle Laboratories for bench testing and seat refurbishment. The valve lifts were performed using water heated to 120°F and pressurized by nitrogen. Valve stem lift was directly measured by a linear voltage differential transformer (LVDT) mounted on the valve stem, and the stem displacement was recorded on a strip chart. The lift setting was indicated by a clear peak on the pressure strip chart and confirmed by spindle displacement measured by the LVDT. Two of the valves had some internal parts replaced prior to the tests. Lift points were 2435 and 2384 psig for valve 2-68-563, 2430 and 2432 psig for 2-68-564, and 2390 psig for 2-68-565.

The different lift points of the Sequoyah 2 valves were attributed to differences in determining the time at which the disc began to lift. The LVDT used at Wyle was accurate to 0.001 inch. The licensee's in situ method used a change in the slope of a trace of lift force on a strip chart, backed up by technicians' confirmation of audible flow.

The licensee found that the setpoint adjustments made as a result of the Sequoyah 2 in situ tests brought the setpoint of the valves outside the limits of the TS. This situation could have resulted in the premature lifting of the safety valves (LER 50/328-88/016).

The licensee for Diablo Canyon has decided to drain the pressurizer safety valve loop seal before testing. The licensee for Sequoyah is planning to send pressurizer safety valves to Wyle Laboratories for future testing.

Hydraulic assist devices have been shown to be accurate in testing spring-actuated safety valve setpoints with saturated steam as the lift medium. They have not been shown to be accurate for safety valve setpoint testing using water or two-phase flow. The results of the Diablo Canyon 1 and Sequoyah 2 tests appear to show that these devices produce inaccurate results when testing pressurizer safety valves with filled loop seals. If the inaccurate results are believed and the valves are reset, a situation can occur in which the setpoints of the valves are low, resulting in valve leakage, and/or premature lift.

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

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Mary S. Wegner, AEOD
(301) 492-7818

Charles G. Hammer, NRR
(301) 492-0919

Attachment: List of Recently Issued NRC Information Notices

LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
88-67	PWR Auxiliary Feedwater Pump Turbine Overspeed Trip Failure	8/22/88	All holders of OLs or CPs for nuclear power reactors.
88-66	Industrial Radiography Inspection and Enforcement	8/22/88	All NRC industrial radiography licensees.
88-65	Inadvertent Drainages of Spent Fuel Pools	8/18/88	All holders of OLs or CPs for nuclear power reactors and fuel storage facilities.
88-64	Reporting Fires in Nuclear Process Systems at Nuclear Power Plants	8/18/88	All holders of OLs or CPs for nuclear power reactors.
88-63	High Radiation Hazards from Irradiated Incore Detectors and Cables	8/15/88	All holders of OLs or CPs for nuclear power reactors, research reactors and test reactors.
88-62	Recent Findings Concerning Implementation of Quality Assurance Programs by Suppliers of Transport Packages	8/12/88	All holders of NRC quality assurance program approval for radioactive material packages.
88-61	Control Room Habitability - Recent Reviews of Operating Experience	8/11/88	All holders of OLs or CPs for nuclear power reactors.
88-60	Inadequate Design and Installation of Watertight Penetration Seals	8/11/88	All holders of OLs or CPs for nuclear power reactors.
88-04, Supplement 1	Inadequate Qualification and Documentation of Fire Barrier Penetration Seals	8/9/88	All holders of OLs or CPs for nuclear power reactors.
88-59	Main Steam Isolation Valve Guide Rail Failure at Waterford Unit 3	8/9/88	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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Although not asked to review the details of the sequoyah event, Frank McCoy read the description and it was discussed with staff on 6/17/88 by Richardson

E. Rossi

*SEE PREVIOUS CONCURRENCES

	D/DOEA:NRR	*C/OGCB:DOEA:NRR	*RPB:ARM	*ROAB:DSP:AEOD
	CERossi	CHBerlinger	TechEd	MSWegner
	08/17/88	08/10/88	07/11/88	06/27/88
*OGCB:DOEA:NRR	*EMEB:DEST:NRR	*C/EMEB:DEST:NRR	*EAD/DEST:NRR	*D/DEST:NRR
BMann	CGHammer	LMarsh	JRichardson	LCShao
06/27/88	06/28/88	07/06/88	07/06/88	07/08/88

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D/DOEA:NRR	C/OGCB:DOEA:NRR	RPB:ARM	C/ROAB:DSP:AEOD	ROAB:DSP:AEOD
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