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

August 22, 1985

IE INFORMATION NOTICE NO. 85-72: UNCONTROLLED LEAKAGE OF REACTOR COOLANT OUTSIDE CONTAINMENT

Addressees:

All boiling water reactors holding an operating license (OL) or a construction permit (CP).

Purpose:

This information notice is provided to alert recipients of a significant event involving an uncontrolled primary coolant leak outside containment. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar events from occurring at their facilities. However, suggestions contained in this information notice do not constitute requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On June 12, 1985, a reactor scram occurred from 99% power at Oyster Creek Nuclear Generating Station. The scram occurred following failure of the electric pressure regulator that subsequently caused a turbine bypass valve to open. This resulted in a reactor pressure decrease to the low pressure trip set point, causing the main steam isolation valves (MSIVs) to shut and the reactor to scram.

As part of the scram sequence, the scram discharge volume (SDV) vent and drain valves are required to shut to contain the water released during a scram. However, in this event, the two drain valves did not fully close, allowing hot, reactor coolant to drain to the reactor building equipment drain tank. The hot fluid flashed in the drain system creating steam that flowed up through various drains in the 51-ft and 23-ft building levels. The steam combined with the fumes from the blistering paint on the SDV drain piping and caused a portion of the reactor building deluge fire protection system to actuate and spray down the 51-ft level. Approximately 500 gal of reactor coolant flowed to the drain tank before the scram system could be reset, which took approximately 38 minutes. The fire protection deluge system actuated approximately 20 minutes after the scram and was shut off in approximately 5 minutes. No safety equipment inside the reactor building was adversely affected by actuation of the deluge system.

Discussion:

The failure of the SDV drain valves to properly close caused the following:

1. Uncontrolled reactor coolant leakage outside containment.

2. Temperatures of the control rod drive (CRD) seals exceeding the alarm setting.

3. Actuation of the reactor building fire protection deluge system.

4. Radioactive contamination of the 23-ft level of the reactor building.

Each SDV drain valve that failed had a different failure mechanism. The upstream valve stem/disc travel stopped approximately 1/8 inch before fully seating onto the valve seat. This was caused by the valve actuator not having the stroking length properly adjusted. The downstream valve had an improperly sized spring in the valve actuator. It is believed that the valve initially closed, but was then forced open when the system pressure exerted a force below the valve seat that exceeded the spring closing force of the actuator.

The high CRD seal temperature alarms were received intermittently after the scram. The alarms are an indication of abnormal flow of reactor coolant within or out of the CRD system. Degradation, or possibly failure, of the seals could occur following prolonged exposure at elevated temperatures. As a result, abnormal leakage could occur that might adversely affect proper rod motion or rod scramming ability.

The reactor building fire protection system actuated on the 51-ft level of the reactor building. Although no equipment was adversely affected by the deluge system spray the potential existed for damaging electrical equipment and possibly aggravating an already serious problem.

Although the SDV vent and drain valves are stroke tested monthly in accordance with the inservice testing (IST) program, there were no criteria or requirements, for leak testing these valves. Following the initial installation of the downstream valve as part of a system backfit in 1984, no postinstallation leak rate test of either valve against operating pressure was conducted. Both valve problems could have been detected by such a test.

IE Information Notice 84-35, "BWR Post-Scram Drywell Pressurization" described an event of August 1982 at the Hatch Nuclear Plant Unit 2 where there was a similar leakage from the SDV. That event was also the subject of an AEOD case study and was included in the 3rd quarter, 1983, "Report to Congress on Abnormal Occurrences."

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

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

Technical Contact: David Powell, IE
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Attachment: List of Recently Issued Information Notices

Attachment 1
IN 85-72
August 22, 1985

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
85-71	Containment Integrated Leak Rate Tests	8/22/85	All power reactor facilities holding an OL or CP
85-70	Teletherapy Unit Full Calibration And Qualified Expert Requirements (10 CFR 35.23 And 10 CFR 35.24)	8/15/85	All material licensees
85-69	Recent Felony Conviction For Cheating On Reactor Operator Requalification Tests	8/15/85	All power reactor facilities holding an OL or CP
85-68	Diesel Generator Failure At Calvert Cliffs Nuclear Station Unit 1	8/14/85	All power reactor facilities holding an OL or CP
85-42 Rev. 1	Loose Phosphor In Panasonic 800 Series Badge Thermo-luminescent Dosimeter (TLD) Elements	8/12/85	Materials and fuel cycle licensees
85-67	Valve-Shaft-To-Actuator Key May Fall Out Of Place When Mounted Below Horizontal Axis	8/8/85	All power reactor facilities holding an OL or CP
85-66	Discrepancies Between As-Built Construction Drawings And Equipment Installations	8/7/85	All power reactor facilities holding an OL or CP
85-65	Crack Growth In Steam Generator Girth Welds	7/31/85	All PWR facilities holding an OL or CP
85-64	BBC Brown Boveri Low-Voltage K-Line Circuit Breakers, With Deficient Overcurrent Trip Devices Models OD-4 and 5	7/26/85	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit