UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

August 10, 2001

NRC INFORMATION NOTICE 2001-13: INADEQUATE STANDBY LIQUID CONTROL SYSTEM RELIEF VALVE MARGIN

Addressees

All holders of operating licenses for boiling water reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addresses to a recent staff finding regarding inadequate standby liquid control system relief valve margin. 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 actions or written response is required.

Background

The control rod drive (CRD) system provides the primary means to control reactivity, as required by 10 CFR Part 50, Appendix A. In the original plant design of Susquehanna Units 1 and 2, the standby liquid control system (SLC) was provided as an independent and diverse (from the CRD system) method for shutting down the reactor under conditions of normal operation. Its specific function was to provide the capability to inject into the reactor a neutron-absorbing solution that was capable of achieving and maintaining subcriticality. At Susquehanna Units 1 and 2, the system included two redundant pumps, each capable of performing the design function.

In 1984 the NRC issued 10 CFR Section 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light water-cooled nuclear power plants" (the ATWS rule). The ATWS rule added more stringent injection rate requirements for the SLC system and required the ATWS functions to be performed under conditions of anticipated operational occurrences. Specifically, the rule required that each boiling water reactor have a SLC system with the capability of injecting into the reactor pressure vessel (during anticipated operational occurrences) a borated water solution at a flow rate such that the resulting reactivity control was at least equivalent to that resulting from the injection of 86 gallons per minute (gpm) of 13 weight percent sodium pentaborate decahydrate (boron) solution.

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Description of Circumstances

To comply with the ATWS rule, the Susquehanna licensee implemented a modification which revised the SLC pump start logic to a simultaneous initiation of both pumps. This resulted in a flow rate of at least 82.4 gpm and a corresponding required concentration of boron solution of 13.6 weight percent. This boron concentration became the licensing basis and was subsequently included in the Susquehanna improved technical specifications.

The change to the pump start logic caused a significant increase in system pressure losses in the pump discharge lines. These losses were the result of the increased fluid velocity in the common injection line as the flow rate doubled from 41.2 gpm to 82.4 gpm. As a result of the ATWS modification, the licensee determined that the maximum discharge pressure at the SLC pumps was 1276 psig. This value was based on the lowest setpoint (1076 psig) of the main steam safety relief valves (SRVs) in the pressure relief mode, the system friction losses for two-pump operation, and the elevation losses. Subsequently, in 1993, the licensee determined a new maximum SLC pump discharge pressure of 1319 psig, based on a power uprate modification. The change was due to a 30 psig increase to the SRV setpoint, and an increase in calculated core flow. The licensee determined that the calculated value of 1319 psig was acceptable because it maintained a 75 psig design margin requirement between the maximum SLC pump discharge pressure and the minimum setting of the SLC pump discharge relief valves (1400 psig).

During a recent design inspection at Susquehanna, the NRC found that the licensee's assumption for reactor vessel pressure used in the maximum pump discharge pressure calculation was non-conservative and disagreed with a vendor ATWS analysis for two of the transients analyzed. Specifically, the inspection team found that for the main steam isolation valve (MSIV) closure transient, the analysis indicated that, at the time of SLC system manual initiation, the reactor vessel pressure would be as high as 1133 psig. Similarly, for the loss of offsite power (LOOP) transient, the reactor pressure at various times in the event was a nominal 1200 psig. The much higher pressure calculated for the LOOP transient event was due to the loss of power to the containment instrument gas compressors and the resulting loss of gas required to open the SRVs. Although each SRV was equipped with a gas accumulator, the amount of gas available in each accumulator was sufficient for only a few SRV actuations. Therefore, the SRV would eventually lift on its higher spring setting (safety mode) and not in its normal pressure relief mode.

Based on the above, the inspection team concluded that the maximum reactor vessel pressure of 1106 psig assumed by the licensee in the design calculations of record was non-conservative. The increases in main steam SRV lift pressure setpoints through the years due to valve simmering concerns and power uprate considerations contributed to the loss of adequate margin between maximum expected pump discharge pressures and the system relief valve settings. This resulted in the likelihood that the SLC pump discharge relief valves would lift during at least one of the ATWS transient scenarios, the loss of offsite power. The lifting of the SLC pump discharge relief valves would cause the sodium pentaborate solution to be recycled to the pump suction and, therefore, prevent the system from meeting the equivalent flow capacity required by the ATWS rule.

Discussion

The licensee modified the Susquehanna Unit 2 SLC system, during a recent refueling outage. The modification increased the flange pressure rating of both pumps (from 1400 psig to 1500 psig) and raised the lift pressure of the pump discharge relief valves to 1500 psig. The licensee intends to perform the same modification on the Unit 1 system. Additional details regarding the issue identified during the inspection can be found in Inspection Report 05000387/01-004;05000388/01-004, Accession # ML011420068.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/ Patrick M. Madden FOR

Ledyard B. Marsh, Chief
Operational Experience
and Non-Power Reactors Branch
Division of Regulatory Improvement Program
Office of Nuclear Reactor Regulation

Technical contacts: Frank Arner, DRS
(610) 337-5194
E-mail: fja@drs.gov

John Richmond, DRP
(570) 542-2134
E-mail: jer4@drp.gov

Neil Della Greca, DRS
(610) 337-5046
E-mail: ald1@drs.gov

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(610) 337-5194
E-mail: fja@drs.gov

John Richmond, DRP
(570) 542-2134
E-mail: jer4@drp.gov

Neil Della Greca, DRS
(610) 337-5046
E-mail: ald1@drs.gov

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