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

November 26, 1993

NRC INFORMATION NOTICE 93-89: POTENTIAL PROBLEMS WITH BWR LEVEL INSTRUMENTATION BACKFILL MODIFICATIONS

Addressees

All holders of operating licenses or construction permits for boiling water reactors (BWRs).

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to potential problems that have been identified by licensees involving hardware modification to the reactor vessel water level instrumentation system. 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

NRC Bulletin (NRCB) 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," issued on May 28, 1993, requested that licensees implement hardware modifications necessary to ensure the level instrumentation system design is of high functional reliability for long-term operation. In response to this bulletin, all BWR licensees with the exception of Big Rock Point, which does not use cold reference leg instrumentation, have either implemented modifications or have committed to implement modifications. The majority of these licensees have decided to install a reference leg backfill system to supply a continuous flow of water from the control rod drive (CRD) hydraulic system through the reference legs to preclude migration of dissolved noncondensible gases down the legs. In August 1993, a potential problem was found at the Susquehanna nuclear power plant during the design of this backfill modification.

Discussion

It was postulated at Susquehanna that a manual isolation valve in one of the reference legs (see Figure 1) could be closed by operator error. Closure of this valve would result in pressurization of that reference leg to CRD system pressure and erroneous indication of low reactor water level and high reactor pressure on all instrumentation associated with that reference leg. The transient resulting from pressurization of the most limiting reference leg includes reactor scram and opening of all safety relief valves (SRVs) due to the false high reactor pressure. The SRVs would remain open and depressurize the reactor until the valves are closed by operator action, or actual reactor pressure falls below approximately 446 kPa [50 psig] at which time the valves can no longer stay open. Reactor depressurization and loss of inventory through the SRVs, in combination with the false low water level signal on the affected reference leg, would result in closure of the main steam isolation valves, actuation of high-pressure and low-pressure emergency core cooling system (ECCS) and containment isolation. Low-pressure ECCS injection would commence after the low-pressure permissive is satisfied. This permissive would be satisfied in this scenario, allowing the low pressure ECCS injection valves to open, because only one of the pressure transmitters is affected and the logic would still be satisfied. A single failure could defeat this logic, however, preventing all low-pressure ECCS injection. The low-pressure permissive can be bypassed in the control room to open the injection valves for all four low pressure core spray (LPCS) pumps. The Susquehanna licensee has informed the NRC that it has physically disabled the manual isolation valves to prevent misoperation of these valves; in addition, the valves are not readily accessible as they are located 6.1 meters [20 feet] above the floor.

This event was recently analyzed for the LaSalle plant by Commonwealth Edison. The analysis indicates that the low-pressure permissive for opening the low-pressure ECCS injection valve would be defeated for the LaSalle design due to the false high pressure signal, thus preventing ECCS injection from the affected division. If a single failure is assumed in the relay for the low-pressure permissive on the other division, no low-pressure ECCS injection would be available. Because the induced plant transient is potentially so severe, LaSalle has designed its backfill modification to make the injection point for the backfill system on the reactor side of the manual isolation valve and excess flow check valve, thereby precluding the potential for pressurization of the reference leg through the backfill system.

Commonwealth Edison took a different design approach for its Dresden and Quad Cities plants. The backfill system design for Dresden and Quad Cities injects into the reference leg on the instrument rack side of the manual isolation valve and excess flow check valve. Additional administrative controls were developed to ensure that the isolation valve would not be inadvertently closed. The licensee analyzed the inadvertent closure of the manual isolation valve for the Dresden and Quad Cities plants and concluded that, while multiple SRVs would open, the resultant plant transient could be mitigated by appropriate operator actions. Without operator actions, the low-pressure ECCS would be available for event mitigation; however, a single failure in the instrumentation system could defeat the low-pressure permissive for opening the low-pressure ECCS injection valves and result in no low-pressure ECCS being available for this transient. The licensee also determined that this design presented an unreviewed safety question because it would increase the probability of a previously analyzed accident, and submitted an application to amend its license pursuant to 10 CFR 50.90. The NRC is currently reviewing the licensee submittal.

Other minor problems with the backfill system have been encountered when installing the system and returning the instrumentation to service after installation was complete. At the Perry plant, a problem occurred when the licensee was in the process of venting one of the instrument lines following the installation of the modification. The job plan directed the operation of the wrong valve, and instead of opening the vent valve the technician opened the isolation valve, allowing air into the reference leg. As a result, the instrumentation associated with the high pressure core spray system (HPCS) was inoperable until it was re-filled and vented. Similar events have occurred at other plants due to procedural inadequacy or lack of attention to detail.

Related Generic Communications

- NRC Information Notice 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," July 24, 1992.
- Generic Letter 92-04, "Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in BWRs Pursuant to 10 CFR 50.54(f)," August 19, 1992.
- NRC Information Notice 93-27, "Level Instrumentation Inaccuracies Observed During Normal Plant Depressurization," April 8, 1993.
- NRC Bulletin 93-03, "Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs," May 28, 1993.

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

Brian Grimes

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contact: Amy Cubbage, NRR
(301) 504-2875

Attachments:
1. Simplified Sketch of Backfill Modification
2. List of Recently Issued NRC Information Notices

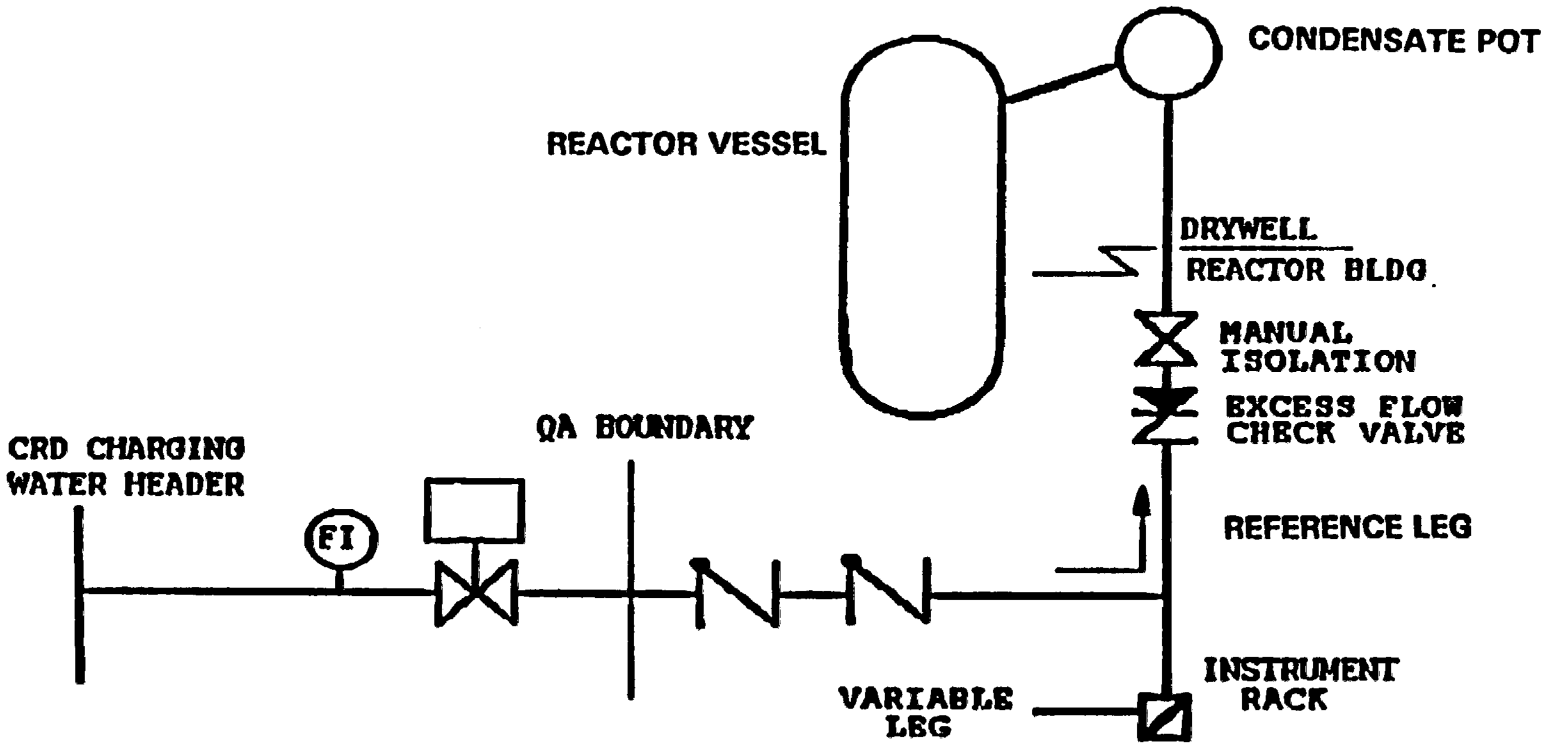

FIGURE 1 - SIMPLIFIED SKETCH OF BACKFILL MODIFICATION

LIST OF RECENTLY ISSUED NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
93-88	Status of Motor-Operated Valve Performance Prediction Program by the Electric Power Research Institute	11/30/93	All holders of OLs or CPs for nuclear power reactors.
93-87	Fuse Problems with Westinghouse 7300 Printed Circuit Cards	11/04/93	All holders of OLs or CPs for nuclear power reactors.
93-86	Identification of Isotopes in the Production and Shipment of Byproduct Material at Non-power Reactors	10/29/93	All holders of OLs or CPs for test and research reactors.
93-85	Problems with X-Relays in DB- and DHB-Type Circuit Breakers Manufactured by Westinghouse	10/20/93	All holders of OLs or CPs for nuclear power reactors.
93-84	Determination of Westinghouse Reactor Coolant Pump Seal Failure	10/20/93	All holders of OLs or CPs for pressurized water reactors (PWRs).
93-83	Potential Loss of Spent Fuel Pool Cooling Following A Loss of Coolant Accident (LOCA)	10/07/93	All holders of OLs or CPs for boiling water reactors (BWRs).
93-82	Recent Fuel and Core Performance Problems in Operating Reactors	10/12/93	All holders of OLs or CPs for nuclear power reactors and all NRC-approved fuel suppliers.
93-81	Implementation of Engineering Expertise on Shift	10/12/93	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

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orig /s/'d by BKGrimes

Brian K. Grimes, Director
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Office of Nuclear Reactor Regulation

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* SEE PREVIOUS CONCURRENCE

SRXB:DSSA*	OGCB:DORS*	TECH ED.*	SRXB:DSSA*	SRXB:DSSA*	D:DSSA*	OGCB:DORS*
ACUBBAGE	PWEN	RSANDERS	WLYON	RJONES	ATHADANI	GMARCUS
11/15/93	11/16/93	11/15/93	11/16/93	11/18/93	11/20/93	11/23/93

D:DORS
BGRIMES
11/26/93
DOCUMENT NAME: 93-89.IN

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EDITED BY: R. Sanders
DATED: 11/15/93

AT

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