

SSINS No.: 6835
IN 87-50

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

October 9, 1987

NRC INFORMATION NOTICE NO. 87-50: POTENTIAL LOCA AT HIGH- AND LOW-PRESSURE
INTERFACES FROM FIRE DAMAGE

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This information notice is provided to alert recipients to a potentially significant safety problem pertaining to the possible initiation of a loss-of-coolant accident (LOCA) as a result of fire damage in the control room or the cable spreading room. If the postulated fire causes a hot short which opens a high pressure to low-pressure system interface isolation valve, exposure of the low-pressure system to pressures in excess of its design pressure could result in a LOCA. It is expected that recipients will review the information for applicability to their facilities and consider actions, if appropriate, to preclude a similar problem. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Background:

The requirements of 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," are applicable to all licensed nuclear power reactor facilities that were operating before January 1, 1979. Facilities that were licensed after that date either committed to comply with the requirements of Appendix R or were reviewed for conformance with the guidelines of the Standard Review Plan (NUREG-0800), Section 9.5.1, "Fire Protection Program," which incorporates the requirements of Appendix R as guidelines. Thus, the same criteria have been used on all nuclear power reactor facilities. In either case, they are simply referred to as the criteria of Appendix R for the purpose of this information notice.

Appendix R states, in part, that where adequate fire protection of safe shutdown systems cannot be maintained, an alternative method of safely shutting down the plant shall be provided. For most plants, an alternate shutdown method is required in the event of a postulated fire in the control room or the cable spreading room. Appendix R also states that for these areas, "...the fission product boundary integrity shall not be affected, i.e., there shall be

8710060172

871009
PDR I&E Notice 87-050

no...rupture of any primary coolant boundary." Thus, for those low-pressure systems that connect to the reactor coolant system (a high-pressure system), at least one isolation valve must remain closed despite any damage that may be caused by fire. A fire could occur in the panel or cables that control the isolation valves causing hot shorts that may result in opening the valves at the high/low-pressure interface. Since the low-pressure system could be designed for pressures as low as 200 to 400 psi, the high pressure from the reactor coolant system (approximately 1000 to 1200 psi for BWRs and 2000 to 2200 psi for PWRs) could result in failure of the low-pressure piping. In many instances, the valves at the high-pressure to low-pressure interface are not designed to close against full reactor coolant system pressure and flow conditions. Thus, spurious valve opening could result in a LOCA that cannot be isolated, even if control of the valve can be reestablished.

Description of Circumstances:

During a fire protection re-review at Washington Public Power Supply System's Washington Nuclear Project Number 2 (WNP-2), the licensee discovered that should a fire occur in the control room, power would have to be removed from the valve motor operators in the residual heat removal (RHR) system suction and discharge lines to prevent inadvertent valve operations resulting from possible fire damage to the circuits. If the damage occurred before removing power to the valve motor operators, the valves could be spuriously opened, resulting in overpressurization of the RHR piping that could lead to a LOCA that could not be isolated.

In discussions with the WNP-2 personnel, the NRC staff became aware of a bypass line around the check valve in the discharge line that had a motor-operated isolation valve in the line. This bypass line is used to warm up the RHR system discharge line by backflow from the reactor before initiating residual heat removal to prevent thermal shocking of the reactor vessel nozzle safe end. Because of this bypass line around the check valve, credit for the check valve in preventing a LOCA at the high- and low-pressure interface can no longer be given.

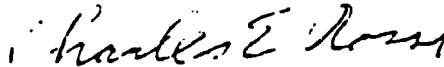
The licensee intends to remove the power to this motor operator during normal power operations. Since this valve is used only for prewarming the RHR line during a normal shutdown, removing power during normal power operations should not adversely impact safe plant operations.

In order to determine if other plants have piping designs similar to that of WNP-2, the final safety analysis reports of nine other BWRs were reviewed by the staff. These included BWR-4, BWR-5, and BWR-6 designs. Of these nine plants, six (Clinton Power Station; Hope Creek Nuclear Station; Limerick Generating Station; Nine Mile Point Nuclear Station, Unit 2; Perry Nuclear Power Plant; and Susquehanna Steam Electric Station) have a piping configuration similar to that of WNP-2. One plant (Monticello Nuclear Generating Plant) has a design similar to WNP-2 but has two normally closed, locally operated manual valves in the bypass line; therefore, this problem does not appear to apply to this plant. The two remaining plants (Grand Gulf Nuclear Station and River Bend Station) do not have bypass lines around the check valve.

IN 87-50
October 9, 1987
Page 3 of 3

The potential for creating a LOCA from a similar high- and low-pressure interface may also be applicable to PWRs.

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


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

Technical Contact: John N. Ridgely, NRR
(301) 492-4742

Attachment: List of Recently Issued NRC Information Notices

Attachment
IN 87-50
October 9, 1987

LIST OF RECENTLY ISSUED
INFORMATION NOTICES 1987

Information Notice No.	Subject	Date of Issuance	Issued to
87-49	Deficiencies in Outside Containment Flooding Protection	10/9/87	All nuclear power reactor facilities holding an O, or OI
87-48	Information Concerning the use of Anaerobic Adhesive Sealants	10/9/87	All nuclear power reactor facilities holding an O, or OI
87-47	Transportation of Radiography Devices	10/5/87	All NRC licensees authorized to manufacture, distribute and/or operate radiographic exposure devices and/or source changers
87-46	Undetected Loss of Reactor Coolant	9/30/87	All PWR facilities holding an O, or OI
87-45	Recent Safety-Related Violations of NRC Requirements by Industrial Radiography Licensees	9/25/87	All NRC licensees authorized to possess and use sealed sources for industrial radiography
87-44	Triple Tube Thinning in Westinghouse Reactors	9/16/87	All PWR facilities employing a nuclear steam supply system holding an O, or OI
87-43	Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks	9/8/87	All nuclear power reactor facilities holding an O, or OI
87-42	Diesel Generator Fuse Contacts	9/4/87	All nuclear power reactor facilities holding an O, or OI
87-41	Failures of Certain Brown Boveri Electric Circuit Breakers	8/31/87	All nuclear power reactor facilities holding an O, or OI

O = Operating License
OI = Construction Permit

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, \$300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
PERMIT No. G 87