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NRCB 88-08

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

June 22, 1988

NRC BULLETIN NO. 88-08: THERMAL STRESSES IN PIPING CONNECTED TO REACTOR COOLANT SYSTEMS

Addressees:

All holders of operating licenses or construction permits for light-water-cooled nuclear power reactors.

Purpose:

The purpose of this bulletin is to request that licensees (1) review their reactor coolant systems (RCSs) to identify any connected, unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses and (2) take action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses.

Description of Circumstances:

On December 9, 1987, while Farley 2 was operating at 33 percent power, the licensee noted increased moisture and radioactivity within containment. The unidentified leak rate was determined to be 0.7 gpm. The source of leakage was a circumferential crack extending through the wall of a short, unisolable section of emergency core cooling system (ECCS) piping that is connected to the cold leg of loop B in the RCS. This section of piping, consisting of a nozzle, two pipe spools, an elbow, and a check valve, is shown in Figure 1. The crack resulted from high-cycle thermal fatigue that was caused by rela-tively cold water leaking through a closed globe valve at a pressure sufficient to open the check valve. The leaking globe valve is in the bypass pipe around the boron injection tank (BIT) as shown in Figure 2. During normal operation this valve

and others isolate the ECCS piping from the discharge pressure of the charging
pumps. With a charging pump running and the valve leaking, temperature stratification occurred in the ECCS pipe as indicated in Figure 1. In addition, temperature fluctuations were found at the location of the failed weld with peak-to-peak amplitudes as large as 70 degrees F and with periods between 2 and 20 minutes. (1)

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(1)/ The staff has learned recently of a problem discovered at Trojan in the pressurizer surge line which involved excessive stresses due to thermal stratification. The staff believes that common elements may exist between the Farley 2 event which necessitated this bulletin and the observations at Trojan. The need for an additional generic communication is being considered as part of our ongoing evaluation of the Trojan event.

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Discussion:

At Farley 2, dual-purpose pumps are used for charging the RCS with coolant from
the chemical and volume control system during normal operation and injecting emergency core coolant at high pressure during a loss-of-coolant accident (LOCA). Separate runs of piping from these pumps are connected to separate nozzles on the RCS piping for normal charging flow, backup charging flow, and hot-and cold-leg ECCS injection and to a nozzle on the pressurizer for auxiliary pressurizer spray. All of these runs of piping, downstream from the
last check valve in each pipe, are susceptible to the kind of failure that occurred in the ECCS piping connected to the cold leg of loop B.

In any light-water-cooled power reactor, thermal fatigue of unisolable piping connected to the RCS can occur when the connected piping is isolated by a leaking block valve, the pressure upstream from the block valve is higher than
RCS pressure, and the temperature upstream is significantly cooler than RCS temperature. Because valves often leak, an unrecognized phenomenon and possibly unanalyzed condition may exist for those reactors that can be subjected to these conditions. Under these conditions, thermal fatigue of the
unisolable piping can result in crack initiation as experienced at Farley 2. Cracking has occurred at other plants in Class 2 systems (see IE Bulletin 79-13, "Cracking in Feedwater System Piping," dated June 25, 1979 and Revisions
1 and 2 dated August 30 and October 16, 1979, respectively). Subjecting flawed
piping to excessive stresses induced by a seismic event, waterhammer, or some other cause conceivably could result in failure of the pipe.

General Design Criterion 14 of Appendix A to Part 50 of Title 10 of the Code of

Federal Regulations requires that the reactor coolant pressure boundary be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. At Farley 2, the pressure boundary failed well within its design life.

Actions Requested:

1. Review systems connected to the RCS to determine whether unisolable sections of piping connected to the RCS can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping. For those addressees who determine that there are no unisolable sections of piping that can be subjected to such stresses, no additional actions are requested except for the report required below.

2. For any unisolable sections of piping connected to the RCS that may have been subjected to excessive thermal stresses, examine non-destructively the welds, heat-affected zones and high stress locations, including geometric discontinuities, in that piping to provide assurance that there are no existing flaws.

3. Plan and implement a program to provide continuing assurance that unisolable sections of all piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the unit. This assurance may be provided by (1) redesigning and modifying these sections of piping to withstand combined stresses caused by various loads including temporal and spatial distributions of temperature resulting from leakage across valve seats, (2) instrumenting this piping to detect adverse temperature distributions and establishing appropriate limits on temperature distributions, or (3) providing means for ensuring that pressure upstream from block valves which might leak is monitored and does not exceed RCS pressure.

4. For operating plants not in extended outages, Action 1 should be completed within 60 days of receipt of this bulletin, and Actions 2 and 3, if required, should be completed before the end of the next refueling outage. If the next refueling outage ends within 90 days after receipt of this bulletin, then Actions 2 and 3 may be completed before the end of the following refueling outage.

   For operating plants in extended outages and for plants under construction, Action 1 should be completed within 60 days of receipt of this bulletin or before achieving criticality, whichever is later, and

Actions 2 and 3 should be completed before achieving criticality, unless criticality is scheduled to occur within 90 days of receipt of this bulletin. In that case, Actions 2 and 3 should be completed before the end of the next refueling outage.

Reporting Requirements:

1. Within 30 days of completion of Action 1, each addressee shall submit a letter confirming that the action has been completed and describing the results of the review. If the review performed under Action 1 indicates that a potential problem exists, the confirmatory letter shall include a schedule for completing Actions 2 and 3.

2. Those addressees who determine that there are unisolable sections of piping that can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping shall submit a letter within 30 days of completion of Actions 2 and 3. This letter should confirm that Actions 2 and 3 have been completed and describe the actions taken.

The written reports, required above, shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended. In addition, a copy shall be submitted to the appropriate Regional Administrator.

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This requirement for information was approved by the Office of Management and Budget under clearance number 3150-0011.

If you have any questions regarding 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: Roger W. Woodruff, NRR
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Attachments:
1. Figure 1 - Farley 2 Temperature Data

2. Figure 2 - Farley 2 ECCS
3. List of Recently Issued NRC Bulletins

Page Last Reviewed/Updated Wednesday, February 16, 2011