OMB No.: 3150-0011
NRCB 88-08, Supplement 3

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

April 11, 1989

NRC BULLETIN NO. 88-08, SUPPLEMENT 3: 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 supplement is to (1) provide information to addressees about an event at a foreign reactor relating to thermal stratification of unisolable piping connected to the reactor coolant system (RCS) similar to the December 9, 1987, Farley 2 event, (2) emphasize to the addressees the need for sufficient review of their RCSs to identify any connected, unisolable piping that could be subjected to unacceptable thermal stratification, and (3) again emphasize the importance of addressees taking action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. No new requirements are included in this supplement.

Description of Circumstances:

On June 6, 1988, while a foreign reactor plant was operating at 100% power, an abnormally high flow rate to the containment sump was detected at about 0.2 gallon per minute (gpm). The source of the leakage was a circumferential crack extending through the wall of an unisolable section of the residual heat removal (RHR) piping that is connected to the hot leg of loop A in the RCS. The crack was 3.8 inches in length on the pipe inner surface and was located on the top of a pipe-to-elbow weld in the horizontal pipe section upstream of the first isolation valve of the RHR suction line as shown in Figure 1. The 8-inch-diameter (schedule 140, 0.8 inch thick) RHR line was fabricated from 316 austenitic stainless steel and was insulated. The crack initiated at the weld-metal-to-base-metal interface on the elbow side and propagated through the weld metal. No material or welding defects were found. Further examination revealed another circumferential crack, which was 4.5 inches in length on the inner surface of the pipe and 60% through-wall, located on the top of the pipe-to-valve weld in the weld metal on the pipe side.

8904050515

The motor-operated wedge-type RHR isolation valve was normally closed. The fluid in the RHR piping upstream of the valve was stagnant and was cool compared to the fluid in the RCS primary loop. There was no leakage from upstream to downstream of the valve. However, the packing gland had been leaking as evidenced by the wet and rusted valve leak-off piping. Subsequent examination of the surface of the lantern rings in the valve packing revealed markings which confirmed leakage from the packing gland. Although the temperature of the leak-off piping was monitored, elevated temperature was not detected because the temperature sensor and the leak-off line were not insulated and were cooled by the environment. The crack resulted from thermal fatigue caused by hot water, which was drawn periodically from the RCS hot leg, leaking through the packing gland of the RHR valve. The hot fluid flowed on top of the cool fluid in the pipe and produced a temperature difference between the top and bottom of the pipe resulting in thermal stresses on the pipe.

The thermal stresses were cyclic because the subject RHR valve permitted the following sequence of events as shown in Figure 2. The stagnant fluid upstream of the RHR isolation valve cooled due to heat loss to the environment. The cool fluid resulted in the thermal contraction of the valve disk. The upstream pressure pressed the disk leaving a small gap between the valve seat and the disk on the upstream side. Fluid flowed through the gap and up to the leaking packing gland and out via the leak-off piping. The leakage drew hot fluid from the hot leg. The hot fluid layer rode on the cool fluid upstream of the valve resulting in thermal stratification. Subsequently, the hot fluid resulted in the thermal expansion of the valve disk closing the gap between the valve seat and the disk stopping the leakage. Then, the fluid upstream of the valve cooled due to heat loss to the environment. The process was repeated continuously, drawing hot fluid from the hot leg and resulting in fatigue due to thermal stratification.

This information was received through discussions with the foreign government and industry representatives.

Discussion:

Although the Farley 2 and foreign reactor events are similar in that they involved thermal stratification in unisolable piping due to valve leakage, there are the following important differences:

(1) Farley 2 involved a small amount of "cold" fluid flow into the RCS, whereas the foreign reactor event involved a small amount of "hot" fluid flow out of the RCS.

(2) Farley 2 involved valve leakage through the valve from upstream to downstream, whereas the foreign reactor event involved valve leakage out of the packing gland.

(3) Farley 2 involved a steady valve leakage which caused the cycling of a check valve due to fluid pressure buildup, whereas the foreign reactor event involved a periodic valve leakage through the packing gland.

The staff at the foreign reactor evaluated several approaches in addressing the concern raised by the event. Attachment 1 describes three approaches that were considered.

Actions Requested:

Although the actions requested in NRC Bulletin 88-08 are unchanged, it should be noted that periodic valve seat leakage through packing glands could result in unacceptable thermal stresses.

Reporting Requirements:

The reporting requirements set forth in NRC Bulletin 88-08 remain unchanged.

If you have any questions about this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate regional office.

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

Technical Contacts: W. Jensen, NRR
(301) 492-1190

S. Lee, NRR
(301) 492-0943

Attachments:
1. Three Approaches Considered in the Foreign Reactor Event Evaluation
2. Fig. 1, "Schematic of the RHR Suction Line at the Foreign Reactor"
3. Fig. 2, "Formation and Fluctuation Mechanism of Thermal Stratification"
4. List of Recently Issued NRC Bulletins

THREE APPROACHES CONSIDERED IN THE FOREIGN REACTOR EVENT EVALUATION

(1) The piping layout may be made such as to minimize the effects of thermal stratification. If the valve is sufficiently far away from the elbow, the high temperature associated with the small amount of leakage flow may dissipate before the leakage flow reaches the valve. Thus, the potential for thermal expansion of the valve disk caused by the leakage flow is minimized. However, because it was difficult to elongate the horizontal pipe due to the limitation of space, the plant staff did not adopt such a change.

(2) The closure of the subject RHR isolation valve was controlled by limit switches which moved the valve disk to a preselected position. With the valve closed, a small gap existed between the valve seat and the disk permitting the cyclic fatigue phenomenon. This small gap may be eliminated by closing the valve to a preselected torque so that the disk is set tightly in the valve seat. However, in order to ensure the valve opening on demand, the plant staff elected not to adopt the approach.

(3) The action selected by the plant staff was to adjust the valve limit switches to position the disk so as to increase the gap between the valve seat and the disk at valve closure. Although the packing gland was replaced a continuous leakage path would exist through the increased gap should the packing gland leakage develop again. By permitting continuous leakage, the cyclic fatigue phenomenon is eliminated. Furthermore, the plant staff insulated the leak-off piping and the associated temperature sensor to provide leakage detection and instrumented the RHR piping with temperature sensors to monitor for thermal stratification.

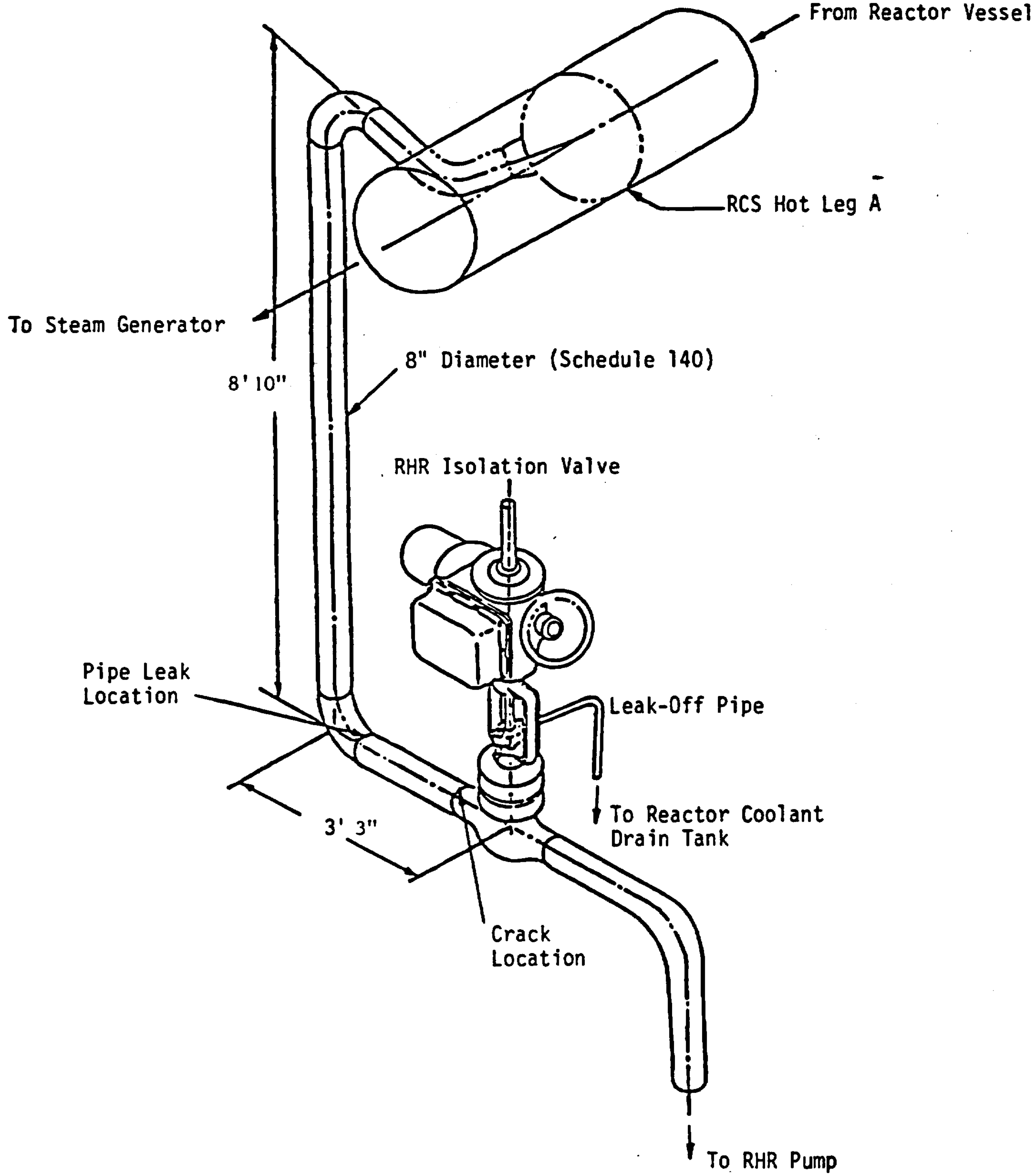


Figure 1 Schematic of the RHR Suction Line at the Foreign Reactor

Attachment 3



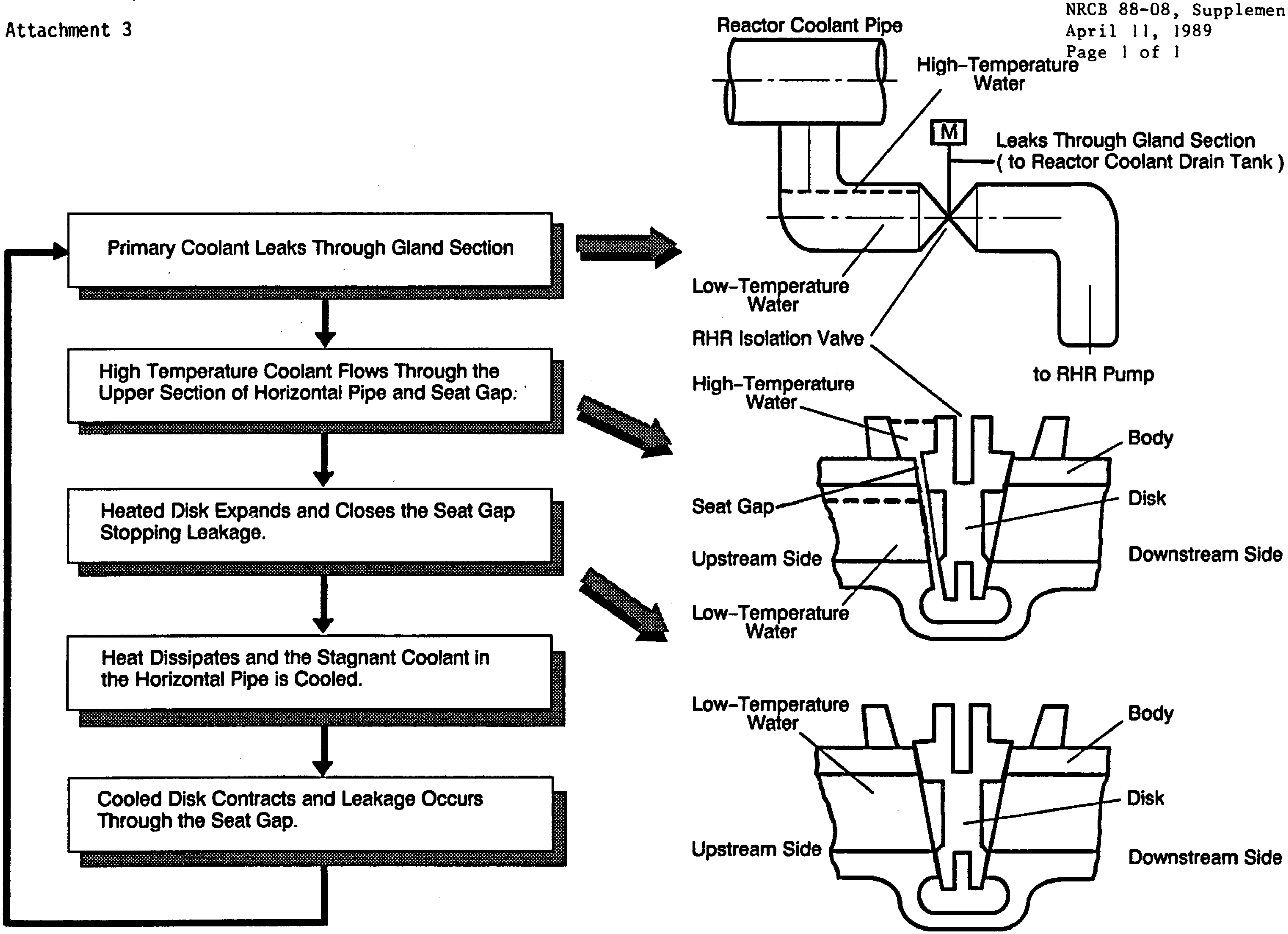


**Figure 2 Formation and Fluctuation Mechanism of Thermal Stratification**

LIST OF RECENTLY ISSUED
NRC BULLETINS

| Bulletin No. | Subject | Date of Issuance | Issued to |
|---|---|---|---|
| 88-07, Supplement 1 | Power Oscillations in Boiling Water Reactors | 12/30/88 | All holders of OLs or CPs for BWRs. |
| 88-11 | Pressurizer Surge Line Thermal Stratification | 12/20/88 | All holders of OLs or CPs for PWRs. |
| 88-10 | Nonconforming Molded-Case Circuit Breakers | 11/22/88 | All holders of OLs or CPs for nuclear power reactors. |
| 88-05, Supplement 2 | Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey | 8/3/88 | All holders of OLs or CPs for nuclear power reactors. |
| 88-08, Supplement 2 | Thermal Stresses in Piping Connected to Reactor Coolant Systems | 8/4/88 | All holders of OLs or CPs for light-water-cooled nuclear power reactors. |
| 88-09 | Thimble Tube Thinning in Westinghouse Reactors | 7/26/88 | All holders of OLs or CPs for W-designed nuclear power reactors that utilize bottom mounted instrumentation. |
| 88-08, Supplement 1 | Thermal Stresses in Piping Connected to Reactor Coolant Systems | 6/24/88 | All holders of OLs or CPs for light-water-cooled nuclear power reactors. |
| 88-08 | Thermal Stresses in Piping Connected to Reactor Coolant Systems | 6/22/88 | All holders of OLs or CPs for light-water-cooled nuclear power reactors. |

OL = Operating License
CP = 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-67

120555026886 1 1C01CY1FB11A1
US NRC-OIRM
DIV OF INFO SUP SVCS
MICROGRAPHICS SPECIALIST
RECORDS SERVICES BRANCH
W/FP-1-21
WASHINGTON DC 20555