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M. O. MEDFORD
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(818) 302-1749

October 18, 1988

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Subject: Docket No. 50-206
NRC Bulletin 88-08
San Onofre Nuclear Generating Station
Unit 1

This letter provides the Southern California Edison response for San Onofre Unit 1 to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems." As requested in Action 1 of the bulletin, a piping design review has been completed. This review has confirmed that there are no unisolable sections of piping connected to the reactor coolant system that can be subjected to excessive thermal stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. Based on the review, no further actions (Action 2 and 3 of the bulletin) are required for Unit 1. Provided as an enclosure are the results of the piping design review for San Onofre Unit 1.

Subscribed on this 18th day of Oct., 1988.

Respectfully submitted,

SOUTHERN CALIFORNIA EDISON COMPANY

By: M O Medford

Subscribed and sworn to before me this
18th day of October, 1988,
C. Sally Sebo
Notary Public in and for the County of
Los Angeles, State of California

OFFICIAL SEAL
C. SALLY SEBO
Notary Public–California
LOS ANGELES COUNTY
My Comm. Exp. Apr. 20, 1990

cc: J. B. Martin, Regional Administrator, NRC Region V
F. R. Huey, NRC Senior Resident Inspector, San Onofre Units 1, 2 and 3

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RESPONSE TO NRC BULLETIN 88-08
THERMAL STRESSES IN PIPING CONNECTED
TO REACTOR COOLANT SYSTEMS

September 20, 1988

The following information is provided in response to Action 1 of NRC Bulletin 88-08 as it relates to San Onofre Unit 1.

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 nondestructively 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 Action 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."

SCE Report of Piping Design Review Results

A design review has been performed for all unisolable sections of piping connected to the reactor coolant system (RCS) to determine whether any of this piping could be subjected to excessive thermal stresses from temperature stratification or temperature oscillations that could be induced by leaking valves. Most of this piping clearly cannot be subjected to such stresses, and was eliminated from further consideration, because it does not have the following prerequisite characteristics necessary (but not sufficient) for such stresses to occur:

1. Availability of a sustained source of water at a pressure greater than that of the RCS. The temperature of this water must be significantly colder than that of the RCS. It was concluded that water supplied at temperatures within 100 degrees F of RCS temperature would not create significant thermal stresses.

2. Isolation of this pressure source from the RCS by one or more closed isolation valve(s) (that is (are) presumed to leak for purposes of this review).

The following piping sections, having both of the above characteristics, were reviewed in more detail:

1. Charging to safety injection (SI) (line nos. RCP-2090-2" and SIS-6008-6") to RCS loop A cold leg,

2. Charging to SI (line nos. RCP-2091-2" and SIS-6006-6") to RCS loop B cold leg,

3. Charging to SI (line nos. RCP-2092-2" and SIS-6007-6") to RCS loop C cold leg, and

4. Pressurizer auxiliary spray (line no. VCC-2080-2").

For each of the first three of these lines, charging to SI, there is no check valve between the isolation valve and the RCS. Any leakage past the isolation valve would cause a slow steady flow rather than oscillations of flow and temperature. Additionally, the isolation valve is far away from the RCS (greater than thirty-five pipe diameters versus approximately five pipe diameters at Farley). Any leakage past the isolation valve would warm gradually before reaching the RCS cold leg. It is concluded, therefore, that none of these three charging to SI lines can be subjected to significant thermal stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

For the pressurizer auxiliary spray line, there is a check valve just downstream of the isolation valve. Both the check valve and the isolation valve, however, are very distant (hundreds of pipe diameters) and approximately forty seven feet below the connection to the main pressurizer spray line. Any leakage flow past the check valve, oscillating or not, would warm gradually before reaching the spray line. Furthermore, the water which would be leaking past the isolation valve comes from the outlet of the regenerative heat exchanger, which is normally at a temperature of 490 degrees F. With no leakage, the temperature at the isolation valve/check valve would be at ambient. With any appreciable flowrate, however, the temperature at these valves would eventually increase, further reducing any resultant thermal stresses even further. It is concluded, therefore, that the pressurizer auxiliary spray line cannot be subjected to significant thermal stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

The conclusion of this piping design review is that no unisolable sections of piping connected to the San Onofre Unit 1 RCS can be subjected to excessive (or even significant) thermal stresses from temperature stratification or temperature oscillations that could be induced by leaking valves.

JMartin:033/skn

OMB No.: 3150-0011
NRCB 88-08

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

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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 relatively 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]

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 nondestructively 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.

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

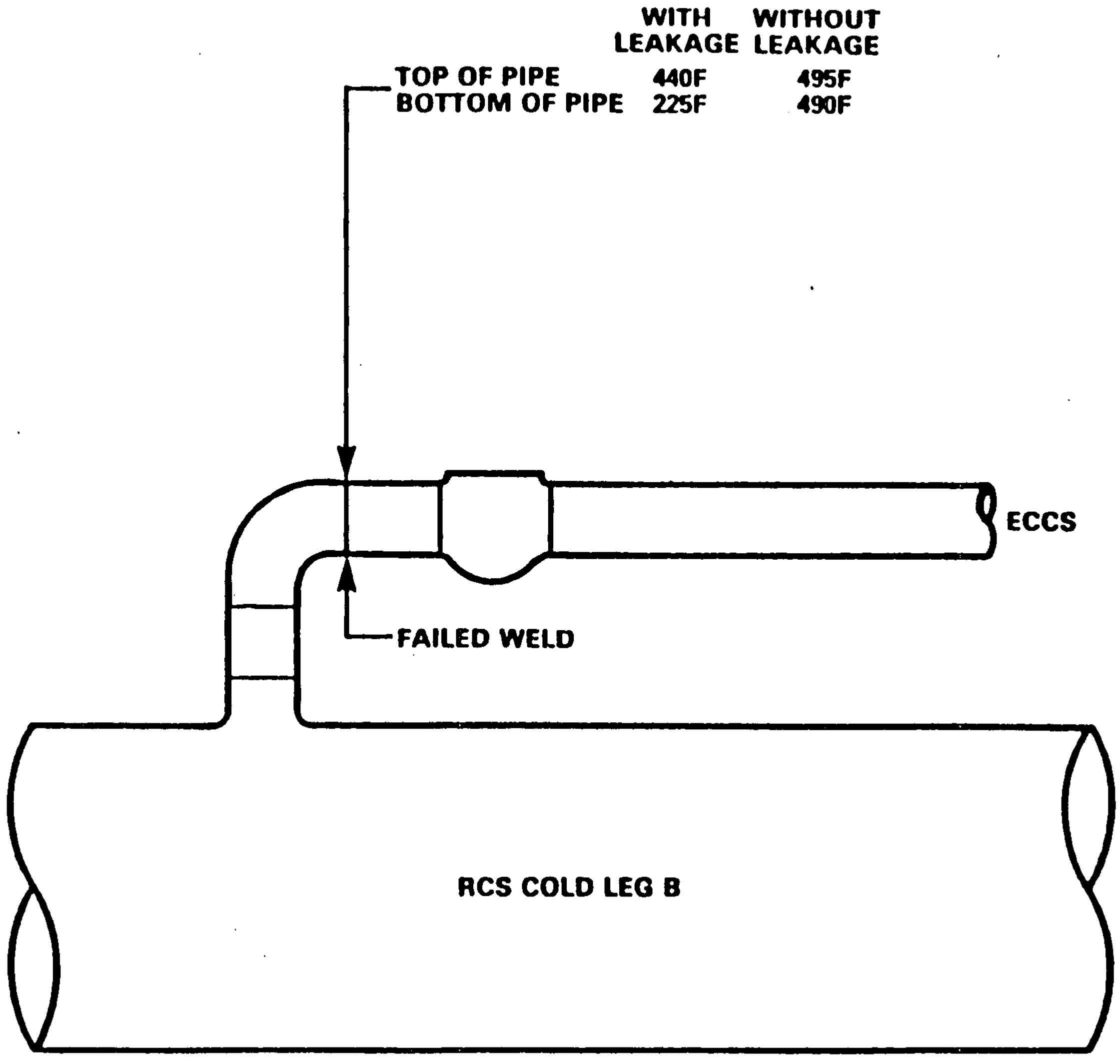
If you have any questions regarding this matter, please contact one of the technical contacts listed below or the Regional Administrator of the appropriate NRC regional office.

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

Technical Contacts: Roger W. Woodruff, NRR
(301) 492-1180

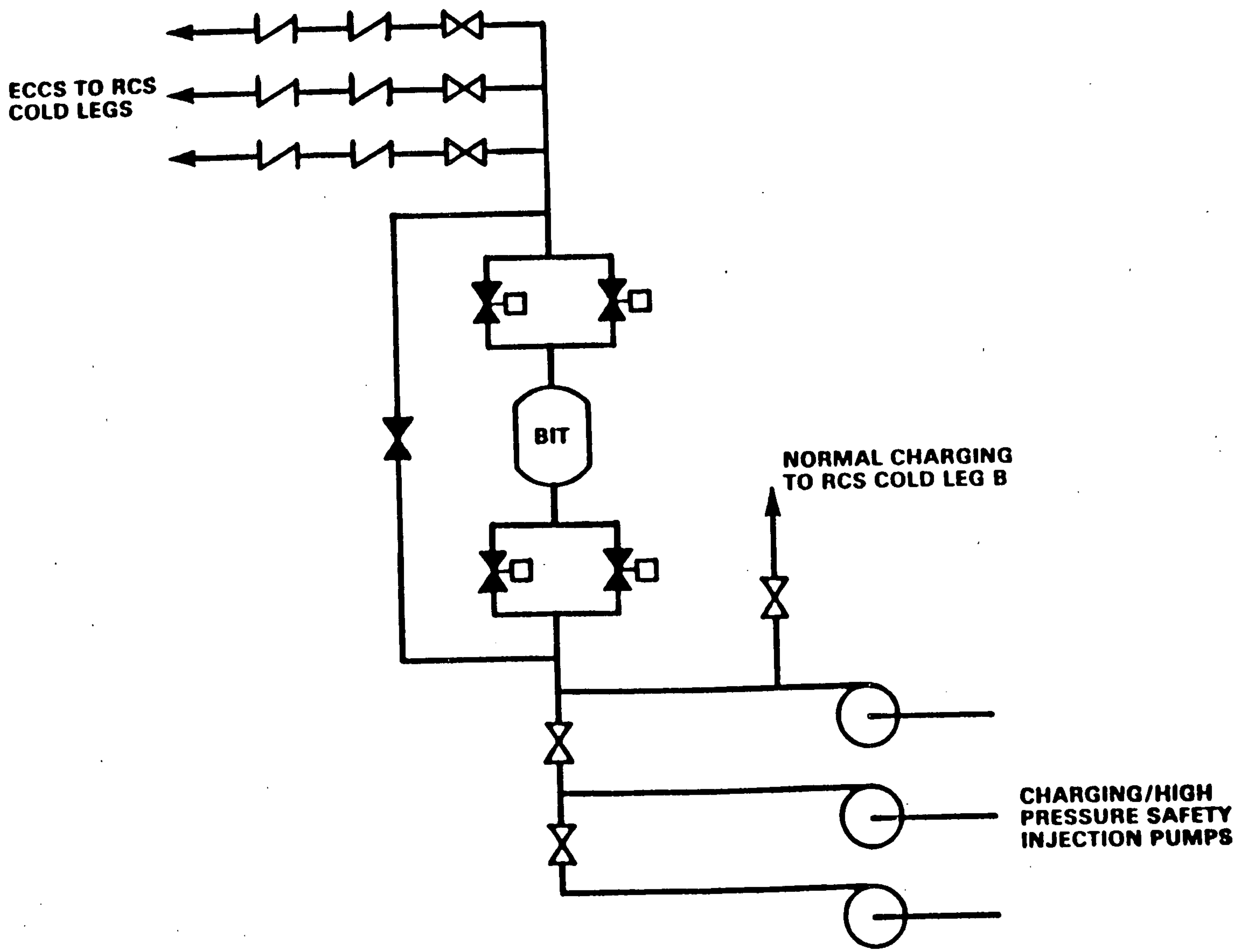
Pao Kuo, NRR
(301) 492-0907

Attachments:
1. Figure 1 - Farley 2 Temperature Data
2. Figure 2 - Farley 2 ECCS
3. List of Recently Issued NRC Bulletins



FARLEY 2 TEMPERATURE DATA

FIGURE 1



FARLEY 2 ECCS

FIGURE 2

OMB No.: 3150-0011
NRCB 88-08, Supplement 1

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

June 24, 1988

NRC BULLETIN NO. 88-08, SUPPLEMENT 1: 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 preliminary information to addressees about an event at Tihange 1 that appears to be similar to the Farley 2 event and 2) emphasize the need for sufficient examinations of unisolable piping connected to the reactor coolant system (RCS) to assure that there are no rejectable crack or flaw indications. No new requirements are included in this supplement.

Description of Circumstances:

Tihange 1 is an 870 MWe, Westinghouse-type, 3-loop, pressurized-water reactor located at Tihange, Belgium. On June 18, 1988, while the reactor was operating, a sudden leak occurred in a short, unisolable section of emergency core cooling system (ECCS) piping that is connected to the hot leg of loop 1 of the RCS. The operator noted increases in radioactivity and moisture within containment and a decrease of water level in the volume control tank. The leak rate was 6 gpm, and the source of leakage was a crack extending through the wall of the piping. The location of the crack and its orientation are shown in Figure 1.

The crack, which is in the base metal of the elbow wall and not in the weld or heat-affected zone, is 3.5 inches long on the inside surface of the elbow and 1.6 inches long on the outside surface. A crack indication also exists in the spool connecting the elbow to the nozzle in the RCS hot leg. That indication is in the heat-affected zone at the weld connecting the spool to the elbow. The indication is circumferential, extends 3.9 inches on the inner surface of the spool, and is 100 mils deep. Two smaller indications exist in the vicinity of the weld connecting the elbow to the check valve.

Farley 2 experienced one crack in a short, unisolable section of ECCS piping connected to an RCS cold leg as described in Information Notice 88-01, "Safety Injection Pipe Failure," and Bulletin 88-08. That crack, which leaked at 0.7 gpm or less, was in the heat-affected zone of the upstream elbow weld. The crack developed slowly rather than suddenly as at Tihange 1.

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Actions Requested:

Although the actions requested in NRC Bulletin 88-08 are unchanged, it should be noted that examinations of high stress locations would include the base metal, as appropriate.

Reporting Requirements:

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

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

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

Technical Contacts: Roger W. Woodruff, NRR
(301) 492-1180

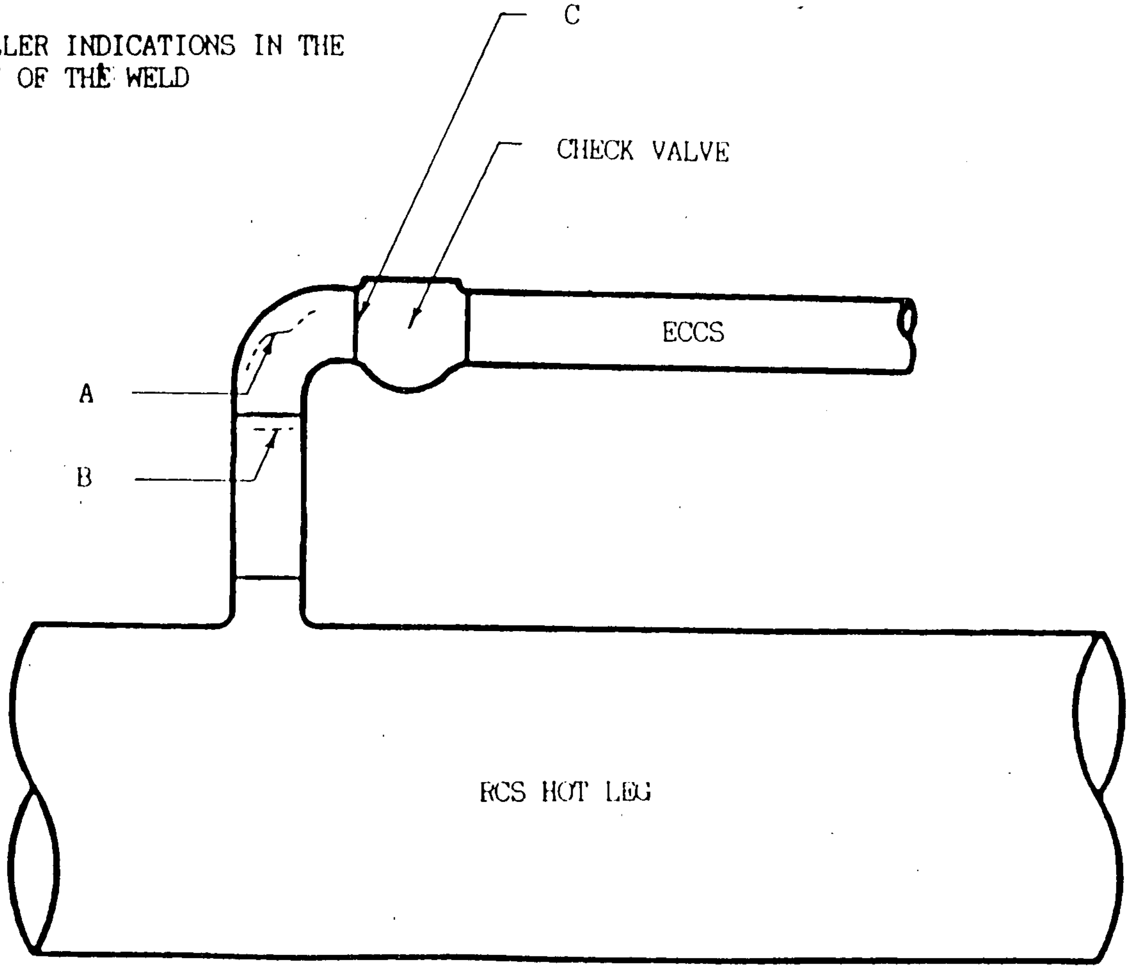
Pao Kuo, NRR
(301) 492-0907

Attachments:
1. Figure 1 - Tihange 1 Piping
2. List of Recently Issued NRC Bulletins

A - THROUGH-WALL CRACK, 3.5 INCHES LONG
INSIDE, 1.6 INCHES LONG OUTSIDE

B - CRACK INDICATION, 3.9 INCHES LONG
INSIDE, 100 MILS DEEP

C - TWO SMALLER INDICATIONS IN THE
VICINITY OF THE WELD



TIHANGE 1 PIPING

OMB No.: 3150-0011
NRCB 88-08, Supplement 2

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

August 4, 1988

NRC BULLETIN NO. 88-08, SUPPLEMENT 2: 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:

This supplement emphasizes the need for enhanced ultrasonic testing (UT) and for experienced examination personnel to detect cracks in stainless steel piping. No new requirements are included in this supplement.

Description of Circumstances:

On the basis of changes in containment atmospheres at Farley 2 and Tihange 1, operators found leakage of reactor coolant from cracks in the first upstream elbow of emergency core coolant system (ECCS) piping connected to the reactor coolant systems. The cracked pipe at both plants was fabricated from 6-inch, type 304, stainless steel components, except for a check valve body at Tihange 1 that was cast, type 316, stainless steel. At Farley 2, the through-wall crack was in the upstream weld and in the heat-affected zones on both sides of the weld. At Tihange 1, the through-wall crack was in the base metal of the elbow. Other cracks at Tihange 1 were found in the pipe spool connected to one side of the elbow and in the body of the check valve connected to the other side. The maximum depth of these cracks was 30 percent of the wall thickness. During repair of the piping, cracks in the check valve body were found by using dye-penetrant testing, and the depth was determined by grinding.

At Farley 2, the weld that failed had been examined on April 17, 1986, as part of the inservice inspection program using the UT technique required by Section XI of the ASME Boiler and Pressure Vessel Code. No reportable flaw indications were found. The same UT procedure was used again after the plant was shut down on December 9, 1987, and again no rejectable flaw indications were reported. After supplementing the UT technique with a 60-degree shear wave transducer and increasing the gain with the 45-degree transducer by 8 db, the through-wall crack was identified. To detect the through-wall crack and other cracks in the Tihange 1 elbow and spool, an instrumentation gain 24 db higher than ASME Code sensitivity was required.

Discussion:

The experience at Farley 2 and Tihange 1 indicates that problems could exist with detection of thermal fatigue cracks in stainless steel piping, fittings, and welds. For the UT procedure to reliably detect these cracks, the practices that were found to provide reliable detection include (1) using sufficient instrument gain so that cracks can be distinguished from non-relevant reflectors, (2) using multiple-angle beam transducers on surfaces that have geometric discontinuities or weld conditions that limit scanning, (3) recording any indication of a suspected flaw regardless of amplitude, and (4) using examination personnel with demonstrated ability to detect and evaluate cracked stainless steel welds.

Personnel training and experience are important considering the elevated scanning sensitivity and the reliance on signal interpretation for reporting and characterizing flaws. The examination procedure describes the acceptance standards and methodology for sizing flaw indications in order to establish actual or conservative flaw dimensions. A UT procedure that has been shown to be capable of detecting and sizing intergranular stress corrosion cracking at boiling water reactors has been demonstrated to be effective in detecting thermal fatigue cracks.

Actions Requested:

Although the actions requested in NRC Bulletin 88-08 are unchanged, reliable examination of stainless steel piping requires specialized UT techniques.

Reporting Requirements:

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

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

Wayne D. Lanning for

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

Technical Contacts: Roger W. Woodruff, NRR
(301) 492-1180

Martin Hum, NRR
(301) 492-0932

Attachment: List of Recently Issued NRC Bulletins

Attachment
NRCB 88-08, Supplement 2
August 4, 1988

LIST OF RECENTLY ISSUED
NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
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.
88-05, Supplement 1	Nonconforming Materials Supplied by Piping Supplies, Inc. at Folsom, New Jersey and West Jersey Manufacturing Company at Williamstown, New Jersey	6/15/88	All holders of OLs or CPs for nuclear power reactors.
88-07	Power Oscillations in Boiling Water Reactors (BWRs)	6/15/88	All holders of OLs or CPs for BWRs.
88-06	Actions to be Taken for the Transportation of Model No. Spec 2-T Radiographic Exposure Device	6/14/88	All NRC licensees authorized to manufacture, distribute, or operate radiographic exposure devices or source changers.
87-02, Supplement 2	Fastener Testing to Determine Conformance with Applicable Material Specifications	6/10/88	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit