

OPERATING EXPERIENCE REGARDING THERMAL FATIGUE OF UNISOLABLE PIPING CONNECTED TO PWR REACTOR COOLANT SYSTEMS

Paul Hirschberg
Arthur F. Deardorff
Structural Integrity Associates
3315 Almaden Expressway, Suite 24
San Jose, CA 95118-1557

John Carey
EPRI Project Manager
1300 Harris Boulevard
Charlotte, NC 28262

ABSTRACT

The U.S. Nuclear Regulatory Commission issued Bulletin 88-08 in response to thermal fatigue failures that occurred at three plants in unisolable portions of piping systems attached to the reactor coolant loop piping. The failure mechanism was thermal stratification caused by valve leakage, and thermal cycling, caused by turbulence penetration. The EPRI Materials Reliability Project (MRP) is currently undertaking a project to develop improved guidelines for screening, evaluation, inspection, and monitoring of thermal fatigue. One of the first steps in developing such guidelines has been to collect operating experience data from domestic PWR plants, to gain an understanding of the quantity and severity of thermal fatigue damage that has actually occurred. By reviewing the failure locations, mechanisms, monitoring results, and corrective actions, more effective tools that address the root causes of actual failures can be developed.

The operating experience data collected consists of two groups: major leak events in domestic and foreign plants, and precursors to leakage, such as thermal monitoring results and observed anomalies in domestic plants. Both sets of data will be available to utilities for review in a database format. The emphasis in this paper will be on the latter group, reviewing the experiences of domestic utilities in implementing monitoring and inspection activities in response to the Bulletin. Each PWR plant was contacted to determine what thermal and other monitoring was implemented, the results, whether leaks or cracks were found, modifications that were made, and unusual occurrences observed that were attributed to thermal fatigue. The scope was limited to the Bulletin 88-08 applicable piping systems. This paper summarizes the results of this survey.

INTRODUCTION

NRC Bulletin 88-08 [1] cited two types of thermal fatigue failures:

- 1) Inleakage events, in which a leaking isolation valve in the high pressure injection system allowed colder fluid to leak into and stratify in the unisolable portion of the reactor coolant system. The upstream pressure of the injection system, which is driven by the charging pumps, is higher than RCS pressure, thus allowing inleakage. Although not fully understood at the time, cracking was caused by turbulence penetration from the RCS flow into the branch pipe, resulting in thermal cycling between the hot RCS flow and cold leakage flow. The events that precipitated the Bulletin occurred at the Farley and Tihange plants.
- 2) Outleakage events, in which intermittent leakage out of the stem leakoff line of the isolation valve in the Residual Heat Removal system allowed hot RCS fluid to leak out of the unisolable portion of the RCS. The intermittent leakage caused stratification and thermal cycling in the unisolable section, resulting in a crack. A leak event at the Genkai plant was the source of the concern.

The Bulletin required all PWR plants to:

- 1) Review all systems connected to the reactor coolant loop that are normally stagnant to identify unisolable sections that are potentially susceptible to cracking from the thermal fatigue mechanisms described in the Bulletin
- 2) In locations that may have been subjected to high thermal stresses, perform nondestructive examinations of the welds, heat affected zones, high stress points, and geometric discontinuities to assure that there are no existing flaws
- 3) Implement a program to assure that thermal fatigue cracking will not occur in these lines by either: a) monitoring the lines to measure thermal stratification and evaluating the results against acceptance criteria, b) preventing pressure upstream of isolation valves from exceeding RCS pressure, or c) installing permanent modifications to enable the system to withstand the stresses.

Since the issuance of the Bulletin, other leakage events have occurred, mostly either in foreign plants, or due to scenarios that were different than those anticipated by the Bulletin. Domestic plants responded to 88-08 in a variety of ways, either instrumenting some or all of the lines with thermocouples, installing pressure monitoring systems, measuring valve leakage, or installing various piping system modifications to preclude thermal fatigue. Insight has been gained from the results of the temperature monitoring performed at these plants. This paper summarizes the results of a comprehensive review of the operating experience related to the Bulletin 88-08 thermal fatigue issue, both from the leak events worldwide, and the domestic monitoring experience and anomalies observed in the course of operation.

DATA COLLECTION METHODOLOGY AND SCOPE

Over 30 references were reviewed to obtain the details of the leak events that have occurred worldwide. This included NRC documents, licensee event reports, conference proceedings, NUREG reports, previously published EPRI reports, operating experience reports, and individual plant failure analyses. Three databases previously compiled by EPRI [2, 3, 4] were searched for relevant items. The NRC's Bibliographic Retrieval System was another source of event information.

The primary source of information for individual plant thermal monitoring experience was a questionnaire that was sent to all domestic PWR plants. The questionnaire solicited information on systems monitored, degree of stratification observed, NDE examination results, leaks and cracks found, other monitoring, and modifications made to reduce thermal fatigue. Responses to the questionnaire were followed up with tele phone conversations to obtain additional details. Other information was obtained from ASME technical papers, stress analyses performed by Structural Integrity Associates, operating experience reports, plant discrepancy reports, and conversations with cognizant plant engineers.

Other thermal fatigue and thermal stratification events have occurred which are beyond the scope of this review. Stratification in the Pressurizer Surge line and Steam Generator Feedwater nozzles are managed by other means and were not included in the scope of the MRP program. Some foreign plant leak events occurred in systems connected to the RCS but were in isolable locations, and were also not included.

RESULTS OF REVIEW - GENERAL

Worldwide, there have been fourteen leak events recorded in unisolable portions of stagnant systems connected to the Reactor Coolant system [5, 6] in PWR plants. Only seven of these were caused by the mechanisms described in Bulletin 88-08. Of these, after the original Farley crack that precipitated the Bulletin, all occurred in foreign plants. From this standpoint, the actions taken by domestic plants in response to the Bulletin were effective in preventing additional failures.

The remaining seven leaks were due to other scenarios not anticipated in the Bulletin but are nevertheless related to the original issue. Four of these pertained to design features that are somewhat unique, and the remaining three had essentially a common root cause. These will be discussed below under the relevant systems.

Domestic plants' actions in response to Bulletin 88-08 varied. Two-thirds of the plants instrumented one or more lines to measure thermal stratification. In most cases, one or two operating cycles were monitored, and if minimal stratification was observed, monitoring was discontinued. One-fourth of the plants performed valve leakage monitoring, either by direct leakage measurement at a test connection, or by applying pressure in a closed section and measuring pressure reduction, or by fluid inventory balancing. In most cases, such monitoring is continuing. Only one plant monitors high pressure injection system pressure upstream of the isolation valve to assure it does not exceed RCS pressure. However, most CE plants, which do not have safety injection systems that operate above RCS pressure, monitor pressure to prevent RCS pressure from entering lower pressure systems due to backleakage through the check valves. About 40% of the plants took some action to mitigate thermal fatigue, such as rerouting piping, adding isolation valves, relocating valves, or improving valve maintenance to prevent leaks.

One of the requirements of the Bulletin was for the plants to review their systems and determine which were susceptible to the thermal fatigue issue. The following were the systems considered by the plants to be the most susceptible and /or required temperature monitoring:

1. Auxiliary Spray
2. High Pressure Safety Injection
3. Alternate Charging
4. Charging / Makeup
5. Intermediate Head (Hot Leg) Injection
6. RHR / Shutdown Cooling / Decay Heat Suction
7. Main Pressurizer Spray
8. Low Pressure Injection
9. Drain Lines / Excess Letdown (after 1995)

RESULTS OF REVIEW BY SYSTEM

High Pressure Safety Injection

This is the system in which the original 88-08 failures occurred, at the Farley and Tihange plants. The system is driven by the charging (or HPI) pumps at pressures higher than RCS pressure. The injection nozzles are smaller diameter than the lower pressure injection nozzles, as low as 1½" on the Westinghouse four loop plants. The lines enter the RCS either from above or from the side.

At Farley, the crack occurred in the heat affected zone of the weld between the first elbow and the horizontal pipe, about 3 feet from the RCS cold leg nozzle, on a 6 inch NPS line. The crack was caused by hot RCS fluid from turbulence penetration interacting at the bottom of the pipe with cold valve leakage fluid that had stratified. The failure was caused by high cycle fatigue, however the current understanding of turbulence penetration is not sufficient to be able to predict the cycling frequency or the depth of penetration with any certainty. The general rule of thumb is that turbulence penetration effects extend between 5 and 25 pipe diameters into the branch pipe.

Four other leak events occurred that were of a similar nature. At Tihange, the crack was located about 2 feet from the RCS hot leg nozzle, in the elbow base metal. At Obrigheim, the crack was located even closer to the RCS, in the nozzle to elbow weld. It had as a contributing cause a deep notch in the circumferential weld that originated in the fabrication process. At Dampierre 2, the crack was in the weld between the check valve and straight pipe just upstream of the hot leg nozzle. At Dampierre 1, the leak occurred in the base metal of the horizontal run between the hot leg and the check valve, about 2 feet from the nozzle. After the failed pipe was replaced, another crack occurred in the same location only nine months later. The replaced section was found to have high residual stresses, and the isolation valve had not stopped leaking.

A somewhat different failure was found at Biblis, where the crack occurred at a tee that connects a hot and a cold injection line. There were contributing causes of a bound up snubber placing high tensile stress on the pipe, and the presence of pump mechanical vibrations. This system design is not found in domestic plants.

It should be noted that none of the safety injection cracks occurred in 1½” nozzles as used in the Westinghouse four loop design. Also, Combustion Engineering plants can be considered as not susceptible to inleakage in the safety injection system because the system is not driven by the higher pressure charging pumps; the highest pressure in the system is about 1500 psi, from the SI pumps. Babcock and Wilcox plants have thermal sleeves in the high pressure injection nozzles, protecting most of the region subject to thermal fatigue.

The results of temperature monitoring in U.S. plants for this system indicated very little thermal stratification during normal operation unless the isolation valve was leaking. The entire pipe from the loop nozzle to the isolation valve generally remained hot. There was only one case where significant stratification was reported: this was due to valve leakage and resulted in a top to bottom temperature difference of 215°F, with cycling having a 2-20 minute period. The valve leakage was corrected and the stratification disappeared.

Although there was little thermal stratification during normal operation, several plants reported stratification during plant heatup for a different reason. During periods when not all reactor coolant pumps are running, a loop with the pump turned off experiences a rise in static pressure. These plants reported that the associated check valve did not seat tightly and RCS flow leaked backwards past the check valve and into the other loops. This resulted in stratification gradients of up to 170°F. This stratification does not pose a concern because it only occurs for a short time and has only a limited number of thermal cycles.

Only one plant installed some form of pressure monitoring on this system. Pressure is checked by a weekly surveillance and is relieved when system pressure exceeds RCS pressure. Two plants installed a second isolation valve to prevent inleakage.

Intermediate Pressure Safety Injection (From SI Pumps)

This system is fed by the safety injection pumps, which operate at about 1500 psi, and usually injects through six inch nozzles in the hot legs. Babcock and Wilcox plants do not have this system, and Westinghouse three and two loop plants use the high pressure injection nozzles for both functions. Combustion Engineering plants use the same nozzle for intermediate head injection, accumulator injection, and low head (shutdown cooling) injection.

A number of plants measured a moderate amount of stratification on these lines but not due to valve leakage. Most of the lines connect to the reactor coolant loop from above, followed by a horizontal pipe run. Top to bottom temperature gradients of up to 100°F were reported in this horizontal section, due to natural convection from the RCS fluid rising to the top of the pipe, and heat loss to the ambient cooling the fluid at the bottom. The stratification appeared to be constant with no cycling, and therefore is not a concern for a thermal fatigue failure.

There were incidents of check valve backleakage, which caused the upstream piping to pressurize to RCS pressure. Most of the CE plants have installed pressure control systems that relieve to the reactor coolant drain tank if backleakage causes the pressure to exceed SI pump pressure. No failures have occurred in these lines.

Low Pressure Safety Injection (RHR and Accumulators)

The low pressure injection nozzles usually perform the combined function of accumulator injection (at about 600 psi) and RHR / Shutdown Cooling injection (at about 350 psi). On B&W plants, Decay Heat Removal / Core Flood injection is done directly into the reactor pressure vessel. The low pressure nozzles are the largest injection nozzles, between 10 and 14 inches in diameter.

No leaks have occurred in these lines. Of the plants that monitored these lines, two reported observing thermal stratification due to natural convection, with no valve leakage and no cycling. These lines tend to have longer horizontal runs with the isolation valve located a longer distance from the loop nozzle. One plant had a check valve leak backwards during heatup, causing stratification in the other lines, when reactor coolant pumps were being cycled. In the B&W plants, only a small amount of stratification occurred in the lines connected to the reactor vessel, with no cycling, as flow in the vessel past the nozzle is insufficient to cause turbulence penetration. None of the above are concerns for thermal fatigue, as there is insufficient cycling to cause a failure.

RHR Suction / Shutdown Cooling / Decay Heat Drop

The RHR Suction line typically drops vertically from the RCS Hot Leg, then travels horizontally for some distance to the isolation valve. Bulletin 88-08 Supplement 3 was issued because of a leak at the Genkai plant. The internals of the isolation valve alternately shrank and expanded causing intermittent outleakage of RCS fluid through the stem packing and leakoff line. The crack, which occurred at the weld between the first elbow downstream of the hot leg nozzle and the horizontal run, was concluded to have been caused by the intermittent leakage periodically introducing hot fluid at the elbow. The horizontal run loses heat to the ambient, thus causing the outleakage to stratify. Had the leakage not been intermittent, there would not have been enough thermal cycles to cause a failure according to the root cause described in the bulletin.

The elbow was 9 pipe diameters from the loop nozzle, well within the range of turbulence penetration from the hot leg flow. In hindsight, it is possible that the cracking was assisted by turbulence penetration causing periodic incursions of hot leg fluid into the horizontal portion of the system, which resulted in the horizontal run periodically stratifying, and the elbow being submitted to cyclic thermal shocks. Most of the responses to the 88-08 Bulletin focused on the valve leakage issue, particularly on stem packing leakage. A number of plants implemented improved valve maintenance programs to reduce packing leaks. Although these programs were effective in stopping packing leaks, they may not have addressed the real root cause of thermal stratification and thermal cycling in the RHR system.

The results of thermal monitoring indicated that numerous plants measured thermal stratification in the RHR suction line without any valve leakage occurring. Top to bottom gradients of up to 350°F were measured at some plants. A more common temperature gradient was 120°F, with the upper temperature cycling +/- 40°, 5-15 cycles per day. The cause of this stratification is turbulence penetration of the hot leg fluid extending into the horizontal pipe run, which stratifies due to natural convection. In order to stratify, the length of the vertical run has to be short enough for the hot fluid to reach the horizontal run, but not so short that the horizontal run is always hot, if the isolation valve is located close to the elbow. A number of plants have long horizontal runs, which increases the propensity for stratification due to heat losses. Some plants noted stratification only during heatup or cooldown, probably because changes in RCS flow during those periods changed the depth of turbulence penetration. One plant noted that during power reductions, the greater the reduction in power, the longer was the turbulence penetration distance. Another plant noted that the turbulence penetration was not cyclic but manifested itself in a random helical pattern.

Charging / Alternate Charging / Makeup

Two leak events occurred in plants having the Babcock and Wilcox Makeup system nozzle design. At Crystal River and Oconee 2, the thermal sleeve became loose and allowed turbulent mixing of hot RCS fluid with cold makeup fluid behind the sleeve. In the B&W design, makeup is not heated by a regenerative heat exchanger and thus the temperature difference between makeup and RCS flow is large. The thermal sleeve was installed with a press fit, and plastic deformation due to hot and cold transients caused it to loosen. The B&W plants have installed an improved thermal sleeve design and implemented a periodic inspection program, such that the problem is considered resolved.

Westinghouse plants typically run either charging or alternate charging, but not both. The concern is that the line that is not in operation could experience inleakage from the charging pump discharge and have the same mechanism for thermal cycling at the high pressure safety injection lines. Of the B&W plants, one uses two nozzles for normal makeup while the others use one. CE plants use both charging paths, so this issue is not a concern for those plants.

The results of thermal monitoring on these lines indicated very little thermal stratification. Either there was no valve leakage, or the distance from the isolation valve to the loop nozzle was short enough that the stagnant fluid did not cool significantly between the regenerative heat exchanger outlet and the loop nozzle.

The B&W plant that uses two makeup nozzles reported stratification of up to 325°F during heatup, due to backflow caused by pressure differences in the two lines when reactor coolant pumps are not running.

Pressurizer Spray (Main and Auxiliary)

The auxiliary spray system in Westinghouse and CE plants draws from the charging system and is stagnant during normal operation. If the isolation valve leaks, charging system pressure can produce flow toward the tee with the main spray system. The fluid in the auxiliary spray system is typically cold because the piping length is long enough to cool to ambient. The main spray comes from the reactor coolant loop cold legs, thus there is the potential for a significant temperature difference between the main and auxiliary spray near the tee, and thermal stratification cycling could result if there is unsteady mixing of the flows from the main spray and the leakage.

The results of temperature monitoring indicated that thermal stratification was minimal during normal operation, unless the isolation valve was leaking. Two plants reported top to bottom temperature gradients of 200°F caused by valve leakage. When the leakage was corrected, the stratification disappeared. Testing and analysis by one plant indicated that the flow velocity of the main spray line was insufficient to cause turbulence penetration to enter the auxiliary spray line. Therefore, although it is possible to have some stratification in the auxiliary spray line from natural convection during normal operation, it is not a concern due to the lack of a cycling mechanism.

During heatup and cooldown, several plants noted significant stratification occurring in these lines. When none or an insufficient number of reactor coolant pumps are running to generate sufficient main spray flow, hot steam from the pressurizer enters the lines. This occurs despite the presence of the “goose neck” loop seal in some of the pressurizer designs. When auxiliary spray was used, a top to bottom temperature gradient of up to 260°F was measured. In addition, auxiliary spray tended to be cycled on and off repeatedly especially during heatups, which results not only in thermal stratification cycles but also thermal shocks at the pressurizer nozzle and auxiliary to main spray tee.

One plant noted significant stratification during letdown isolations. Apparently there was some isolation valve leakage of charging flow, which was significantly cooler due to the lack of letdown flow in the regenerative heat exchanger.

One leak event occurred in Finland at the Loviisa plant. The crack was in a Z-type isolation valve in the auxiliary spray line, vertically above the main spray tee. During heatup and cooldown, pressurizer steam entered the valve outlet, and auxiliary spray entered the inlet; the hot outlet was at a lower elevation than the cold inlet, and stratification cycling occurred internally in the valve. There had also been a preexisting material inclusion defect in the valve that facilitated cracking.

Reactor Coolant Loop Drains / Excess Letdown

The reactor coolant loop drain lines and the excess letdown line were not recognized as being susceptible to developing unisolable leaks due to thermal stratification cycling in any of the plant reviews in response to Bulletin 88-08. However, four leak events have occurred in these lines. At Three Mile Island, a leak occurred in a cold leg drain line, in the weld between the first elbow downstream of the loop nozzle and the horizontal pipe run. At Oconee, a leak happened in almost the exact same location, this time near the center of the elbow extrados. The TMI drain line was 1½” NPS, with an increase to 2” in

the horizontal run; the distance from the inside surface of the reactor coolant loop to the crack was 14". Similarly, at Oconee the drain line was 1½" and the distance to the crack was 13". The crack was caused by turbulence penetration from the RCS intermittently extending into the horizontal run. In both cases the pipe was not insulated, which made it easier for the horizontal portion to stratify. The vertical pipe length was just right for the hot fluid to periodically extend into the horizontal run, but not so short as to keep the horizontal pipe warm all the time.

At Mihama, the same mechanism caused a leak in the excess letdown line. The line was 2" NPS and the crack was located 15" from the reactor coolant loop inside surface. One difference was that the Mihama line was insulated; in this case, however, the length of the horizontal run to the isolation valve was very long, resulting in an equivalent amount of heat loss to ambient as a shorter, uninsulated line. There is also evidence that there was an unsteady stratification layer in the elbow. The Mihama elbow was also found to have had high tensile residual stress induced in the fabrication process. The corrective action for this event was to shorten the vertical run such that the turbulence penetration boundary was well into the horizontal run, away from the elbow, which is a point of stress concentration.

The other leak event occurred at Loviisa, in a cross tie line between a hot leg drain and a cold leg drain. The crack was in a weld between a reducer and the tee that joins the two drains. Leakage past the cross tie valve allowed hot leg fluid to flow into the cold leg drain. Intermittent thermal expansion of the valve internals caused thermal cycling at the tee. This system design is not found in domestic plants.

Since none of the domestic plants initially identified the RCS drains as a potentially susceptible system, none of them performed any temperature monitoring. However, one plant had a leak occur in the excess letdown line in a similar location as in Mihama. In this case the crack was in the heat affected zone of the weld between the first off elbow and the horizontal run, 13" from the reactor coolant loop. The utility concluded that the cause was a thermal interference between a flange on the line and a floor support plate, which generated a stress cycle each time the reactor coolant loop heated up or cooled down. At another plant, a crack was found in a hot leg nuclear sampling system isolation valve. The crack was caused by admitting 600°F samples on a daily basis into a normally stagnant line.

CONCLUSIONS

Thermal stratification and cycling is still a concern for fatigue cracking in lines connected to the reactor coolant loop. However, the susceptible systems can probably be limited to high pressure safety injection, RHR suction, and reactor coolant loop drain lines, or similar lines. More work needs to be done to improve the understanding of turbulence penetration effects, particularly how to determine the penetration distance and the rate of cycling.

The details of the worldwide leak events and the domestic plant monitoring experience are being made available in database format on EPRI WEB. The database will be kept up to date by adding new operating experience as submitted by users.

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