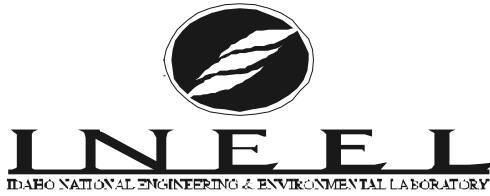# Assessment of Field Experience Related to Pressurized Water Reactor Primary System Leaks

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Assessment of Field Experience Related to
Pressurized Water Reactor Primary System Leaks[a]

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ABSTRACT

This paper presents our assessment of field experience related to pressurized water reactor (PWR) primary system leaks in terms of their number and rates, how aging affects frequency of leak events, the safety significance of such leaks, industry efforts to reduce leaks, and effectiveness of current leak detection systems. We have reviewed the licensee event reports to identify the events that took place during 1985 to the third quarter of 1996, and reviewed related technical literature and visited PWR plants to analyze these events. Our assessment shows that USNRC licensees have taken effective actions to reduce the number of leak events.   One main reason for this decreasing trend was the elimination of reportable leakages from valve stem packing after 1991. Our review of leak events related to vibratory fatigue reveals a statistically significant decreasing trend with age (years of operation), but not in calendar time. Our assessment of worldwide data on leakage caused by thermal fatigue cracking is that the fatigue of aging piping is a safety significant issue. Our review of leak events has identified several susceptible sites in piping having high safety significance; but the inspection of some of these sites is not required by the ASME Code. These sites may be included in the risk-informed inspection programs.

INTRODUCTION

This paper presents our assessment of the U.S. experience relating to pressurized water reactor (PWR) primary system leaks in terms of their number and rates, how aging affects frequency of leakage events, the safety significance of such leakages, industry efforts to reduce the leak-age events, and effectiveness of current leak detection methods. Five specific actions were taken to perform the assessment: (a) review of licensee event reports (LERs) related to leak events, (b) development of a database to identify trends, distributions, and causes of leak events, (c) visits to PWR plants, (d) review of related technical literature including USNRC communications and reports prepared and/or submitted by licensees, and (e) analysis of selected leak events.

The scope of our study was to review the LERs relating to PWR primary system leaks submitted during the period 1985 through the third quarter of 1996, representing about 638 operating years for U.S. PWRs. Several leak events that took place outside the study period or, in the case of thermal fatigue, outside the United States were reviewed to complement this study. The review included only those leak events that occurred during hot shutdown, hot standby, startup, and power operation. Leak events that took place during cold shutdown and refueling were not included, nor were the events associated with intersystem leaks or steam generator tube leaks.

We present our specific findings in four areas: (1) trends of annual rates of U.S. PWR primary coolant leaks, (2) trends in world-wide leak events caused by thermal fatigue, (3) safety significance of piping thermal fatigue, and (4) information relevant to risk-informed inspection. Other findings related to risk significance of the leak events and effectiveness of leak detection systems may be found in the NUREG report by Shah et al. (1998).

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TRENDS OF ANNUAL RATES OF PWR PRIMARY COOLANT LEAKS

We searched the USNRC Sequence Coding and Search System (SCSS) database to identify the LERs that are potentially associated with the reactor coolant leak events. We reviewed the LERs located by this search and identified 215 LERs associated with leak events that occurred during the study period. Some of the 215 LERs were associated with more than one leak event, that is, a leak occurred at more than one location in the RCS. As a result, we found 240 leak events associated with the 215 LERs included in the study. Of the 240 events, 199 leak events (associated with 174 LERs) were reported because of leaks. The leak rate in most of these events exceeded a plant technical specification limit, and were reportable leaks in accordance with Title 10 of the Code of Federal Regulations, Part 50, Section 50.73 (10 CFR 50.73), "Licensee Event Report System." In some of the 199 events, the leak rate was smaller than the technical specification limit, but the event was reported because there was a potential for the leak to exceed the limit. Several leak events associated with reactor coolant pump seal degradation fall into the later category. Hereafter, these 199 leak events are referred to as reportable leak events.

The remaining 41 events were instances where a reactor coolant leak was mentioned in the LER narrative, but the LER was not prepared because of leak. These 41 events represent nonreportable leak events and are not included in the trends presented here.

Primary System Leak Event Frequencies and Leak Rates

The trend of the 199 reportable leak events is shown in Figure 1. More than half of the leak events (121 events) occurred during the first 4 years of the study period. The frequency of leak events is also shown in Figure 1, and is calculated by dividing the number of reportable leak events for a given calendar year by the number of PWR operating years

Figure 1. Distribution of reportable leak events and their frequencies during the 1985–1996 period. The frequency for 1996 is estimated based on the operating years for 1995, accounting for data only through the third quarter of 1996.

for the same year. The frequency of reportable leak events has significantly decreased since 1986. One reason for the decreasing trend of leak events is the elimination of valve stem packing degradation as a major mechanism causing reportable leak events.

Investigation of the data presented in Figure 1 shows a statistically significant decreasing trend in the frequencies of the reportable leak events as presented in Figure 2. The solid line shows the estimated trend and the dotted lines show 90% confidence bands. For this investigation, the trend, if present, is assumed to have an exponential form. The data are barely consistent with this modeling assumption because two of the thirteen 90% confidence intervals do not overlap the fitted line. A formal goodness-of-fit test did not quite reject the assumed exponential form. Excluding the first 4 years from the analysis, the reportable leak events that occurred since 1988 do not show any statistically valid trend.

Distribution of the reactor coolant leak rates for 153 events, which occurred inside the containment, is as follows. The maximum leak rate was less than 3.8 L/min (1 gpm) for about 29% of the events, whereas it was greater than 76 L/min (20 gpm) for about 7.2% of the events. The maximum leak rates were not reported for about 25% of the events. The highest leak rate was 760 L/min (200 gpm), which resulted from a transient-induced LOCA event that occurred at Fort Calhoun on March 7, 1992. A pressure transient caused a pressurizer safety valve to lift.
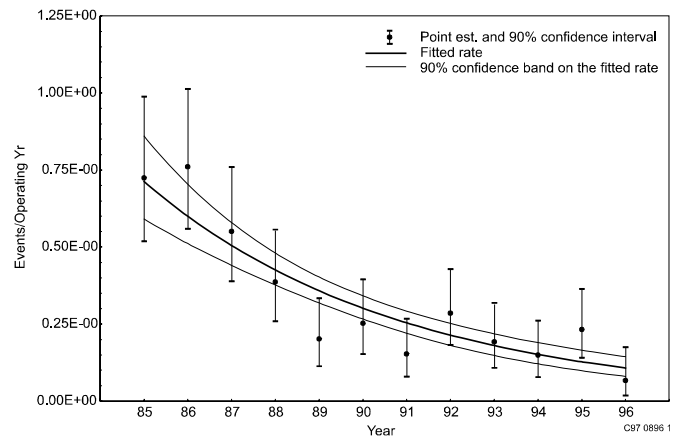

Figure 2. Statistical analysis of a trend in the frequency of reportable leak events showing point estimates, 90% confidence bands, and 90%

confidence intervals.

Safety injection was demanded and subsequently the plant had to depressurize and cool down. The second highest leak rate was 494 L/min (130 gpm), which was caused by a failed compression fitting on the RCS hot leg instrument line at Oconee 3 on November 23, 1991. This was an isolable leak, but could not be isolated until the plant was placed in cold shutdown and airborne activity levels decreased sufficiently to allow containment entry.

Out of 199 reportable leaks, there were 67 unisolable leaks; 45 of which were pressure boundary leaks. Seven leaks were isolable leaks but the environment resulting from the leaks made it difficult to isolate them. An isolable leak is defined as a leak that can be stopped by closing a valve or placing a valve on its backseat. A leak that was isolable with difficulties could have been isolated because of plant design; however, because of a personnel hazard associated with such a leak, it could not be stopped until the plant changed operating modes. The frequency of pressure boundary leaks showed no statistically significant trend during the study period.

The distribution of the maximum reactor coolant leak rates for the 67 unisolable leak events is as follows: leak rate less than 3.8 L/min for 34 events; between 3.8 L/min and 5 L/min for 12 events; between 5 and 10 gpm for 3 events; and 25, 40, and 200 L/min for one event each. The leak rate was not known for the remaining 15 events.

Dominant Causes of the Primary System Leaks

Degradation mechanisms were identified for 124 leak events. The three major degradation mechanisms causing primary system leaks are vibratory fatigue (29 events), packing degradation (29 events), and stress corrosion cracking (SCC) (16 events) (Shah et al. 1998). Other mechanisms include, for example, seal and gasket degradation. Leak events associated with packing degradation show a significant decreasing trend with none reported since 1991. Leak events associated with the other two major mechanisms do not show any trend in calendar time. However, the leak events associated with vibratory fatigue do show a decreasing trend in age (time in service); this fact is presented later in this section.

The distribution of the 16 leak events associated with SCC by the type of mechanism is as follows: ten events were caused by primary water stress corrosion cracking (PWSCC), 3 by transgranular stress corrosion cracking (TGSCC), 2 by intergranular stress corrosion cracking (IGSCC), and the SCC mechanism is not known for one event. For the 10 leak events caused by PWSCC, leakage occurred through the Alloy 600 base metal in eight events and through the weld metal used with Alloy 600 components in one event; the location for the 10th event was not reported. The maximum leak rate associated with these events is ≤1.5 L/min (0.4 gpm).

The two components with the most leaks were the valves and pipes, 97 for valves and 42 for piping and instrument lines. The highest leak rate from a valve was 760 L/min (200 gpm) at Fort Calhoun as discussed. The next two highest leak rates, 179 and 171 L/min (47 and 45 gpm),

were caused by packing degradation and occurred in 1987 and 1991, respectively. The highest leak rate in the piping, 331 L/min (87 gpm), was caused by vibratory fatigue and occurred in 1986.

The distribution of the 42 leak events in piping and instrumentation lines by degradation mechanism is presented in Table 1. Out of 42 leaks, 40 were through welds in stainless steel piping and instrument lines, and 2 were through the Alloy 600 base metal and were caused by PWSCC.

Table 1. Distribution of reportable leak events in piping and instrument lines by degradation mechanisms.

Degradation Mechanism	Pipe Size, mm		
	≤25 mm	between 25 and 102 mm	between 102 and 305 mm
Vibratory Fatigue	15	13	
Thermal Fatigue		1	1
Mech. Fatigue		1	
PWSCC	2		
IGSCC		1	1
TGSCC	1		
SCC	1		
Unknown	4	1	

Trends in Leak Events Caused by Vibratory Fatigue Cracking

Trends of the vibratory fatigue-related leak events in calendar time and plant age were statistically investigated as shown in Figures 3 and 4, respectively. The ef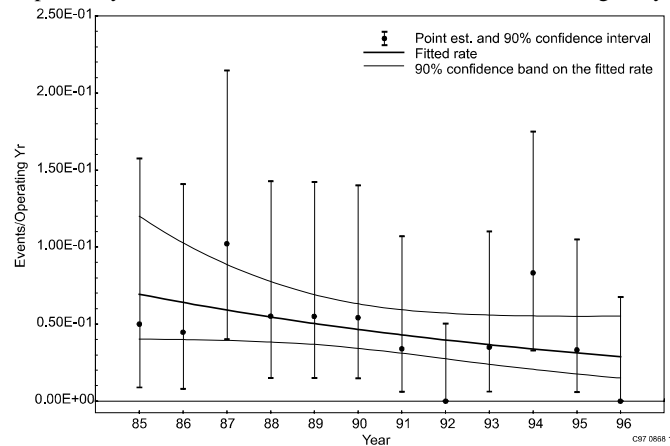fect of calendar time reflects the evolving body of regulations, design improvements, and industry-wide learning. The effect of plant age reflects the learning of plant personnel and the aging of the hardware. Solid curves in Figures 3 and 4 show the estimated trend in calendar time and with age, respectively. The dotted lines show 90% confidence bands. A trend in this investigation, if present, is assumed to have an exponential form. The data are consistent with this modeling assumption, because all point estimates of 90% confidence intervals overlap the fitted trend line. The results in Figure 3 reveal no statistically significant trend in calendar time, whereas the results in Figure 4 reveal a statistically significant decreasing trend with age. Apparently, the decreasing trend implies that the vibratory fatigue failures are caused by premature aging because of inadequacy of the initial design and fabrication. In other words, the decreasing trend implies that the vibratory fatigue failures are not caused by aging damage resulting from long-term operation.

TRENDS IN WORLD-WIDE LEAK EVENTS CAUSED BY THERMAL FATIGUE

Thermal fatigue cracking leading to PWR primary system leaks is not widespread in nuclear power plants; only four leaks have been reported in U.S. plants, two of which took place during the study period. Therefore, we have complemented these data with world-wide data for such leak events. The world-wide data are associated with all Western-designed PWRs and two Loviisa units from Finland.
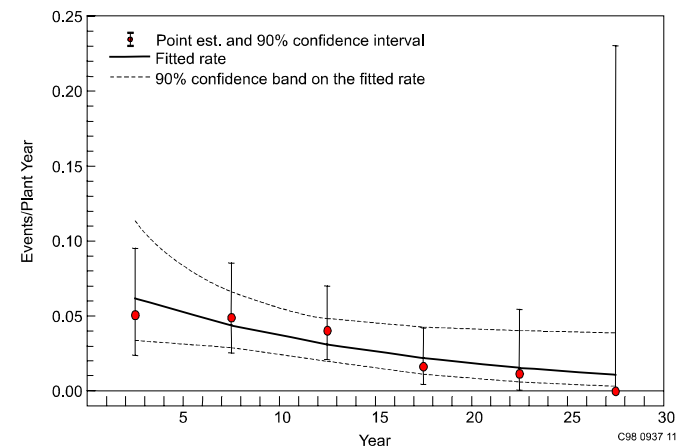

Figure 3. Statistical analysis of a trend in the reportable vibratory fatigue-related leak event frequencies estimated by calendar year. Point estimates, 90% confidence bands, and 90% confidence intervals are shown.

Figure 4. Statistical analysis of a trend in the reportable vibratory fatigue-related leak event frequencies estimated by plant age (years in service). Point estimates, 90% confidence bands, and 90% confidence intervals are shown.

Thirteen leak events caused by thermal fatigue cracking of PWR reactor coolant piping have been reported (Jungclaus et al. 1998). In addition to these events, thermal fatigue cracks have been detected at two EDF plants during 1997: Dampierre 3 and Fessenheim 2 (INES 1997). Most of these thermal fatigue failures occurred because of the turbulent

penetration, thermal cycling, turbulent mixing, and thermal stratification and striping phenomena that were not taken into account in the original design (Shah et al. 1998)

The 13 leak events are listed in Table 2 along with the data related to piping system through which leakage occurred, throughwall crack location and size, and leak rate for each event. These events took place in the U.S., France, Belgium, Finland, Germany and Japan. One event took place during the plant initial startup test, 2 took place during the first 10 years of operation, and 10 took place during the 10-to-25 year period of operation. For most of these events, throughwall cracking was in an unisolable portion of the small-diameter (≤~200 mm) reactor coolant piping. In eight of these events, the throughwall cracking was in the weld or its heat-affected zone. In the other four events the throughwall cracking was away from the weld and in the base metal of an elbow, straight pipes, and a valve body. In the remaining one event, the throughwall cracking was both in the weld and adjacent base metal. The leak rate was ≤ 3.8 L/min. (1 gpm) in 9 events, and ≥3.8 L/min. in the remaining four events. The maximum leak rate was 500 L/min. during the May 1998 leak event at Civaux 1 plant.

Trends of the thermal fatigue-related leak events (listed in Table 2) in terms of plant age were statistically investigated, as shown in Figure 5. We used the world list of nuclear power plants (as of December 1997) published in the March 1998 issue of Nuclear News. All Western-designed PWRs were counted from this list; two Loviisa units from Finland were also included. Both operating reactors and the reactors that have been shutdown were counted, 217 reactors in all. For the reactors that have been shutdown, the years when the reactors were in operation were included in the analysis. The plant ages were collapsed into 5-year groups, as shown in Table 3, and the number of leak events was counted for each group. The ages of the plants were calculated as of May 31, 1998. The zero count for age group 25-30 is not surprising because very few plants are that old. The extrapolated value of estimated leak frequency (leak events per reactor year) for this age group is 1.5E-2. For this frequency, it is more likely to see zero leak events in 43.9 reactor years than to see one or more events in that time period.

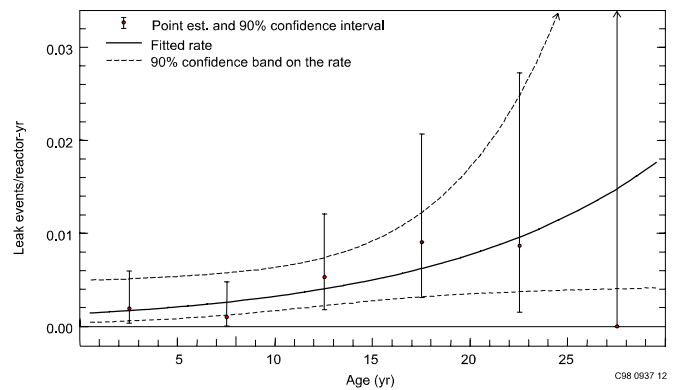
Figure 5. Statistical analysis of a trend in the thermal fatigue-related leak event frequencies estimated by plant age (years in service). Point estimates, 90% confidence intervals, and 90% confidence bands are shown.

Table 2. PWR reactor coolant leak events caused by thermal fatigue.

Plant	Event Date	Initial Criticality Date	Piping System	Throughwall Crack		Leak Rate (L/min.)
				Location	**Size**	
Crystal River 3[1]	1/82	1/77	Makeup/High Pressure Injection	Check valve body near the valve-to-safe end weld	140-degree circumferential crack; two crack initiation sites: one on the inside surface and one on the outside surface	3.8
Obrigheim[2]	/86	9/68	Chemical and Volume Control	Weld between a 90-degree elbow and a nozzle	Crack extended 70 mm circumferentially at the inside surface, 12 mm at the outside surface	0.02
Farley 2 [3]	12/87	5/81	Safety Injection	Heat affected zone of elbow-to-pipe weld	Crack extended 120 degrees circumferentially at the inside surface, 25-mm long at the outside surface	2.7
Tihange 1[4]	6/88	2/75	Safety Injection	Elbow base metal	89-mm long at the inside surface, 41-mm long at the outside surface	22
Genkai 1[5]	6/88	1/75	Residual Heat Removal	Heat-affected zone of elbow-to-pipe weld	Crack extended 97 mm circumferentially at the inside surface, 1.5-mm at the outside surface	0.8
Dampierre 2[6]	9/92	12/80	Safety Injection	Check valve-to-pipe weld and base metal of straight portion of pipe	Crack extended 110 mm circumferentially at the inside surface, 25 mm at the outside surface	10
Loviisa 2[7]	5/94	10/80	Spray Line	Pressurizer auxiliary spray line control valve body	Crack extended 80 mm along the horizontal surface and 25 mm along the vertical surface of the valve body	few drops
Biblis-B[8]	2/95	3/76	Chemical and Volume Control System	Base metal of a straight portion of the pipe	Crack extended 50 mm axially at the inside surface, 20 mm at the outside surface	66.7
Three Mile Island 1[9]	9/95	6/74	Cold Leg Drain Line	Weld between a 90-degree elbow and a 51-mm diameter horizontal line	Crack extended 51 mm circumferentially at the inside surface, 14 mm at the outside surface	0.06
Dampierre 1[10]	12/96	3/80	Safety Injection	Base metal of a straight portion of the pipe	The crack extended 80 mm circumferentially at the inside surface, 22 mm at the outside surface	2.7
Loviisa 2[11]	1/97	10/80	Hot Leg Drain Line	Weld between a T-joint piece and a reducer	65-degree circumferential crack	0.5
Oconee 2[12]	4/97	11/73	Makeup/High Pressure Injection	Safe-end to pipe weld	Crack extended 360 degree circumferentially at the inside surface, about 77 degree circumferentially on the outside surface	45.6
Civaux 1[13, 14]	5/98	initial startup test phase	Residual Heat Removal	longitudinal weld in an elbow	180-mm long throughwall crack	500

[1] Babcock & Wilcox 1983   [2] Jungclaus et al. 1998 [3] Farley 1987
[4] Pirson and Roussel 1998   [5] Shirahama 1998   [6] Jungclaus et al. 1998
[7] Hytonen 1998   [8] Jungclaus et al. 1998 [9] Three Mile Island 1995
[10] Jungclaus et al. 1998   [11] Hytonen 1998   [12] Duke Power 1997
[13] INES 1998   [14] MacLachlan 1998

Table 3. Distribution of thermal fatigue-related PWR RCS leak events by plant age.

Age (years in service)	Reactor Years	Number of Leak Events	Leak Events/Reactor Year (frequency)
0.0-5.0	1052.1	2	1.9E-3
5.0-10.0	982.5	1	1.0E-3
10.0-15.0	756.9	4	5.3E-3
15.0-20.0	442.4	4	9.0E-3
20.0-25.0	230.9	2	8.7E-3
25.0-30.0	43.9	0	0

The dots and vertical bars show point estimates and 90% confidence intervals for the leak frequency, each based only on the data for one 5-year age group. A trend in this investigation, if present, is assumed to have an exponential form. The solid curve in the figure shows the estimated trend with age. The dotted lines show a 90% confidence band. The data are consistent with the exponential modeling assumptions because all the confidence intervals overlap the fitted trend. The results in the figure reveal that the leak events associated with thermal fatigue follow a statistically significant increasing trend with age.

SAFETY SIGNIFICANCE OF PIPING THERMAL FATIGUE

Our assessment of world-wide data for the leak events caused by thermal fatigue indicates that the fatigue of aging PWR branch lines may become an issue as plants get older. Our reasons for the assessment are mainly based on the phenomena causing thermal fatigue, crack morphology and its growth, and limitations of inservice inspection of branch lines.

Thermohydraulic phenomena that caused the thermal fatigue cracking at Farley 2, Tihange 1, and other PWRs are not yet well understood. Qualitative understanding of these phenomena has been developed under a program sponsored by the Electric Power Research Institute (EPRI) to investigate thermal stratification, cycling, and striping (TASCS) (EPRI 1993). However, it appears that the quantitative aspects of turbulent penetration and thermal cycling phenomena have not been fully developed under the TASCS program and significance of these phenomena in the failures at Farley 2 and Tihange 1 has not been clearly established. The TASCS experimental program and analytical methodology do not predict the locations of these failures correctly. The TASCS methodology predicts higher cyclic stresses at the end of the turbulent penetration column where thermal cycling may take place if valve inleakage is present, and lower stresses at locations within the turbulent penetration column where temperature differences approach zero. But when the TASCS methodology is applied to the Farley 2 safety injection line fatigue failure, the throughwall crack location is within the calculated length of the turbulent penetration column and not at the end of the column where the cyclic stresses are expected to be higher and a fatigue failure is more likely to occur. Because the Tihange 1 failure is similar to the Farley 2 failure, the TASCS methodology is not likely to predict correctly the throughwall crack location in the Tihange 1 safety injection line. This discrepancy between the calculated and actual location of cracking implies that the thermodynamic phenomena that caused these failures are not well understood (USNRC 1996, 1997a; Lund and Hartzman 1998).

Past experience with thermal fatigue cracking indicates that the crack growth is slow, and it leads to leakage but does not challenge the structural integrity of the pipe. Experience with a rapidly growing fatigue crack, however, is limited. Such a rapid crack growth occurred at Dampierre 1 in 1997. A portion of the safety injection line was replaced during the repair for the 1996 leakage event (see Table 2). A crack initiated and propagated to 67% throughwall depth in the replaced piping within 8 months after the replacement. This result contradicted the fatigue analysis results for the replaced piping, which indicated that the crack should not initiate for years, even when taking into account local thermal loads revealed by temperature monitoring of the piping (Merle 1998).

The French safety authority recently identified a concern related to analytical evaluation of safety margins for thermal fatigue cracking. The stability of big cracks in a small diameter piping under seismic conditions combined with a cold leakage through a valve is not well assessed by analysis. The earthquake can make the cold leak more severe, resulting in larger thermal loads on piping. Physical tests evaluating the stability of such cracks are not yet performed. An experimental validation of the margin term may be needed (Merle 1998).

Field experience shows that a large leak rate through a thermal fatigue crack could lead to a small break loss-of-coolant accident (SBLOCA) For example, a leakage rate of 500 L/min (132 gpm) through a fatigue crack in residual heat removal (RHR) line was reported at Civaux 1 at the end of its initial startup test phase (INES 1998, MacLachlan 1998). Such throughwall cracking might occur without being detected during inservice inspection because of low crack initiation time, high crack growth rate, and a large crack size. The recent experience at Dampierre 1 indicates that a small, nondetectable fatigue crack in a safety injection line could become a throughwall crack in one fuel cycle, indicating a high crack growth rate.

The recent cracking at Oconee 2 suggests the possibility of uniform crack growth that may lead to a long throughwall crack. The growth of the circumferential crack was uniform (~ 30% through wall) over about 78% of the circumference as shown in Figure 6 (USNRC 1997b). The faster growth in the remaining portion of the crack was probably due to cold fluid flowing up from the warming line, causing thermal mixing in the upper portion of the pipe cross section near the damaged weld.[1] In the absence of the warming line flow, the crack growth would have been more uniform, which could eventually lead to a longer throughwall crack accompanied by a larger leak rate.

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[1] V. N. Shah, private communication with J. M. Davis, Duke Power, November 17, 1998.
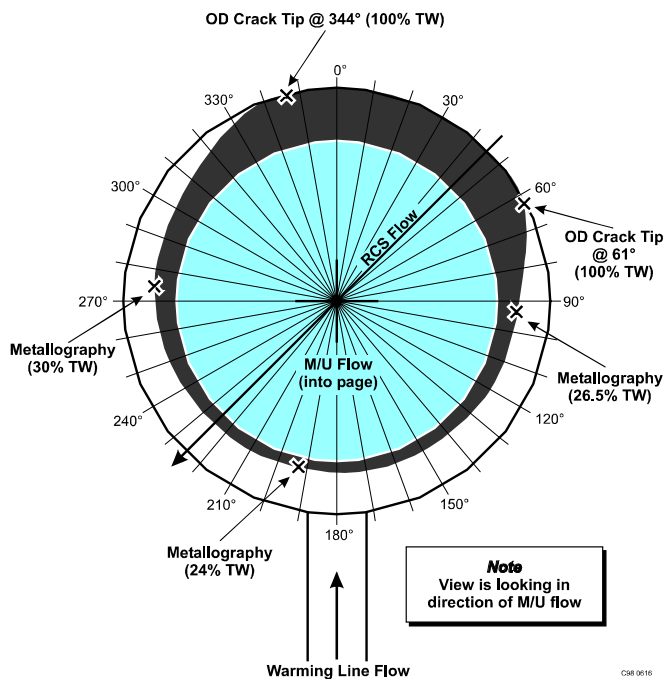
Figure 6. Angular distribution of depth of the circumferential crack at the safe-end-to-MU/HPI line weld.

Applying the leak-before-break concept to PWR piping with a diameter smaller than 152 mm (6 in.) is difficult. The concept has been approved for the main coolant piping at most U.S. PWRs and for several other piping, including residual heat removal and safety injection piping with a diameter equal to or greater than 152 mm, but not for any smaller-diameter (< 152 mm) piping (Wichman et al. 1997).

Current inservice inspection techniques and requirements for branch lines, that is, small diameter piping, have several limitations. ASME Section XI requirements for Class 1 piping with smaller than a 102-mm (4-in.) diameter are not adequate to detect thermal fatigue damage. The requirements include only surface examination of the welds (ASME 1989), but volumetric examinations are needed to detect thermal fatigue cracks, which initiate on the inside surface.

It is difficult to detect thermal fatigue cracks at weld and base metal sites during inservice inspection when the plant is shut down. It is more difficult to size these cracks. These difficulties are more relevant for small diameter piping. For example, 33 to 66% of throughwall cracks were not detected during inservice inspection in France (Merle 1998). Qualified inservice inspection techniques are needed to characterize thermal fatigue cracks in the PWR branch lines.

At present, inspection of susceptible base metal sites (away from welds) is generally not required at U.S. PWRs. But five of the 13 leakages listed in Table 2, which were caused by thermal fatigue cracking

and took place world wide, have occurred through base metal. Recently, because of the 1996 leak event at Dampierre 1, the inservice inspection program for the French PWRs has been revised to include the inspection of base metal of the unisolable portion of the safety injection lines (Gauthier 1998). Shah and Ware (1994) also recommended inspection of base metal sites susceptible to thermal fatigue cracking. Risk-informed inspections may address this inspection need at U.S. PWRs.

Current inservice inspection (ISI) methods may not be effective in detecting a thermal fatigue crack before it becomes through wall if the crack growth rate is high. Recent experience at Dampierre 1 indicates that a small, nondetectable fatigue crack in a safety injection line could become a through-wall crack in one fuel cycle.

INFORMATION RELEVANT TO A RISK-INFORMED INSPECTION PROGRAM

Our review and analysis of leak events has identified several sites in piping that have high safety significance and are susceptible to cracking. These sites may be included in a risk-informed inspection program. Data relevant to risk-informed inspection program are as follows:

- Leakage has occurred from piping with 254-mm (10-in.) or smaller diameter. No leakage has been reported from larger diameter piping.

- Degradation mechanisms that have caused throughwall cracking in piping include vibratory fatigue, thermal and mechanical fatigue, and three stress corrosion cracking mechanisms [PWSCC, TGSCC, and IGSCC].

- Affected piping materials are mainly the 300-series stainless steels. One exception is Alloy 600 material, which is used for penetrations.

- Vibratory fatigue cracking of socket welds has been the predominant cause for leakages from piping. The associated piping has a diameter smaller than 102 mm (4 in.).

- Leakage has generally occurred through piping welds, with two exceptions. In the case of PWSCC, leakage has generally occurred through base metal. In the case of thermal fatigue, 5 of 13 leakages have occurred through base metal.

- Throughwall cracking has generally initiated on the inside surface, with some exceptions. The main exception is vibratory fatigue, which has initiated at the root or toe of socket welds. Mechanical fatigue and TGSCC can also initiate cracking at the outside surface. For example, mechanical fatigue initiated cracking at the outside surface of the weld during the 1982 leak event at Crystal River 2, and TGSCC initiated cracking at the outside surface of a drain line during the 1998 leak event at Oconee 1 (Duke Power 1998). Cracking at the outside surface has been initiated at welds, not in the base metal away from the welds.

- It is essential that qualified inspection techniques that can reliably detect the suspected damage be used. As mentioned earlier, French field experience indicates that the current inspection methods did not detect 33 to 66% of throughwall cracks during inservice inspection.

- It may be difficult to identify all susceptible locations for risk-informed inspection. One option may be to inspect all unisolable welds in branch lines and other small-diameter piping. One U.S. utility is evaluating this option (Duke Power 1998).

- ISI may not be able to detect a crack before it becomes through wall if the crack growth rate is fast. A small defect, nondetectable during regular ISI, could become a throughwall crack within an operating cycle. Generally, vibratory fatigue causes such a rapid crack growth in socket welds. But, as mentioned earlier, thermal fatigue has also caused such a rapid crack growth at Dampierre 1 in 1997. Therefore, other monitoring and inspection programs may be needed to identify the susceptible locations in a timely manner: (1) monitoring of valve leakage and inspection for a loose thermal sleeve can assist in identifying sites susceptible to thermal fatigue cracking, and (2) vibration monitoring can identify socket welds susceptible to vibratory fatigue cracking.

SIGNIFICANT FINDINGS

The paper has presented some of the findings of our assessment of field experience related to PWR primary system leaks. Five significant findings are summarized as follows.

1. Reportable leak events show a statistically significant decreasing trend in calendar time.

2. Thermohydraulic phenomena that caused thermal fatigue cracking in PWR branch lines are not yet well understood.

3. Leaks through thermal fatigue cracks in branch lines could lead to an SBLOCA.

4. Thermal fatigue has caused throughwall cracking in the base metal at several PWRs.

5. Fast growing thermal fatigue cracks may require monitoring of valve leakage and pipe wall temperatures. Such monitoring can also be used to identify other susceptible locations.

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