Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors

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The piping systems in various pressurized water reactors (PWRs) and boiling water reactors (BWRs) within the United States have shown signs of degradation. This degradation has resulted in cracking and leaks in some piping. The U.S. Nuclear Regulatory Commission (NRC), concerned about the cracking and leaks occurring in the operating nuclear power plants within the United States, has formed several pipe crack study groups (PCSG) to address this problem. This paper summarizes the results of a study made by the NRC PCSG concerning the cracking incidences in PWRs through September 1980. The recommendations presented in this paper are those of the PCSG and do not present an official position of the NRC.

1. Introduction

The U.S. Nuclear Regulatory Commission (NRC) is concerned about the cracking and degradation occurring in the various piping systems of operating nuclear power plants within the United States. The staff and their consultants have studied various aspects of degradation to determine the specific causes and establish ways it can be reduced or eliminated.

At the Fifth International Conference on Structural Mechanics in Reactor Technology (SMiRT) 1979, the authors summarized the results of the NRC studies on cracking and degradation through early 1979 [1]. Since that time, additional pipe cracking has been observed in pressurized water reactor (PWR) nuclear plants within the United States. The NRC Pipe Crack Study Group (PCSG) has investigated this problem and published its findings in NUREG-0691 [2]. This paper summarizes the PCSG investigation.

2. Pipe Cracking in PWRs

During the past decade, there have been isolated instances of feedwater line cracking in PWRs. The problem did not become generic, however, until mid-1979 when leaking was observed in the D. C. Cook Unit 2 nuclear plant. Since then, examinations of Combustion Engineering and Westinghouse plants have disclosed 16 incidents of cracking at 35 plants examined. Generally, these incidents of cracking have been attributed to thermal fatigue due to stratification of hot or cold water and periodic injection of cold water, predominantly during hot standby. These incidents of cracking are summarized in Table I.

There have been several instances of cracking and failure of small-diameter lines in PWRs, principally attributed to fatigue. In addition to the repeated failures in small lines, there have been instances of miscellaneous pipe cracking in a spectrum of PWR systems in which no pattern was apparent.

There are two other classes of repetitive occurrences in the PWRs operating in the U.S. The first encompasses those instances of stress corrosion cracking (SCC) in systems containing boric acid, where the maximum damage was limited to leakage. Most such instances occurred in relatively thin-walled piping. The second class of repetitive occurrences has been caused by water hammer or water slugging. Incidents have ranged from little or no damage to breakage or damage to pipe supports, and in a few instances to breakage of piping.

3. Stress Corrosion Cracking in PWRs

No occurrences of IGSCC have been reported to date in piping for PWR primary coolant systems [1]. Furthermore, IGSCC is not likely to occur in primary piping because all the conditions required for IGSCC are not present. In particular, the dissolved oxygen in the coolant is controlled to levels low enough to preclude IGSCC through the use of hydrazine additions during startup and a hydrogen overpressure during operation. Other impurities that might cause SCC, such as halides and caustics, are also rigidly controlled. Only for brief periods when the reactor is shut down and the coolant is exposed to air are conditions considered even marginally capable of producing SCC in the primary system of PWRs.

Although not widespread, IGSCC and transgranular stress corrosion cracking (TGSCC) have occurred in PWR piping in systems other than the primary coolant systems from which oxygen is not rigidly excluded. To date, stress corrosion cracks have been reported in eight PWR units in the United States. Table II lists these instances, along with the systems involved and the pipe sizes.

In PWR cracking, the corrodent has not been firmly identified in all cases. Water with a high oxygen level as well as contaminants of chlorides, flourides, caustics and sulfur compounds are suspected. The cracking has occurred only in piping normally or intermittently containing stagnant water where, because of the lack of flow, concentration cells may be established that accelerate the initiation of IGSCC and/or impurities may become absorbed in the oxide films on the stainless steel or be dissolved from the steel surface. All of the cracks occurred in pipes containing borated (H_3BO_3) water which had some dissolved oxygen and low measured levels of chloride and sulfur compounds.

The generic problem of IGSCC in PWRs is still under investigation. Studies are being funded by the Electric Power Research Institute (EPRI), NRC, and others to investigate the causes and provide remedies to this condition in PWR environments and to improve existing detection methods.

In the past, the problem was corrected on a case-by-case basis. More recently, failed piping has been removed and replaced with a less susceptible material such as a low-carbon stainless steel. Thin-gauge pipe was sometimes replaced with thicker pipe. At TMI Unit 1, repairs were made using a combination of low-carbon stainless steel replacement material in conjunction with a corrosion-resistant cladding of the in-place piping to which the welds of the replacement spool pieces are attached. This approach essentially eliminates any sensitization in the highly-stressed areas as a result of welding, and thus mitigates the susceptibility to IGSCC.

4. Cracking in PWR Feedwater Lines

In May 1979, cracking was reported to have occurred in the D. C. Cook plant feedwater lines in the immediate vicinity of the steam generator (SG) nozzles. After an inspection of all PWR feedwater lines, a number of additional cracks was discovered.

Metallurgical analysis on specimens removed from the affected units showed that the cracks initiated at a machining discontinuity introduced in the piping elbow. Figure 1 is a schematic drawing showing the location of the crack and typical repair configuration. From metallographic and fractographic analysis, the mode of failure appears to be fatigue possibly assisted by corrosion. Analysis of in-plant instrumentation and thermohydraulic modeling indicates that thermal fatigue is the dominant cracking mechanism. Horizontal portions of the feedwater lines were subjected to significant temperature differences between the top and bottom of the pipes during hot standby, startup, and shutdown conditions. This phenomenon is illustrated in Figure 2. The metal surface at the interface between the hot and cold fluid is subjected alternately to hot and cold water which results in cyclic thermal stresses in the pipe. Analysis indicates these stresses can approach the yield strength of the material.

Short-term corrective actions taken involved the replacement or repair of the cracked piping using material similar to that originally used. Care was taken to reduce the presence of the stress concentration by increasing the machined radius (see Figure 1).

Long-term remedial measures are being investigated. These include the possibility of heating the feedwater during low flow conditions or adding a thermal sleeve in the cracked region. Either of these will reduce the thermal gradient in the pipe.

5. Vibration Fatigue Cracking of Small Lines

A review of licensee event reports (LERs) since 1969 indicates that 84 cracks have occurred in small lines. Cracks have been located predominantly in socket welds in the 3/4-in. to 2-in. diameter vent, drain, and instrument pipe lines. All small line breaks were identified by pipe leakage. Most fatigue failures resulted from induced vibrations from pumps in the piping system. However, some fatigue failures have resulted from thermal fatigue and erosion from impinging fluid on the piping wall.

Corrective actions in instances of fatigue and miscellaneous cracking or small lines were based on a case-by-case basis. For vibration-induced fatigue failures, weld repair or replacement has been the predominant corrective action taken. In some instances, lines have been rerouted and dampers have been installed to reduce vibrations.

6. Recommendations

A summary of the PCSG recommendations involving short-term, intermediate, and long-term licensing implementation for CP, OLs, and operating plants is presented below. These recommendations do not present an official position of the NRC.

6.1 Small-diameter lines in PWR secondary systems where breaks occur by fatigue: The generic scoping analyses indicate that the functional capability of certain PWR secondary systems may be degraded by small-diameter line breaks to less than that assumed in the plant Final Safety Analysis Report.

6.1.1 Short-Term Implementation Action - Plant-specific scoping analyses should be performed for all plants to identify PWR secondary systems whose function may be significantly degraded by small-diameter-line breaks. If systems are determined to be degraded, instruct plant operators of appropriate action in the event of a small-diameter-line break in the affected system.

6.1.2 Intermediate Implementation Action - More detailed analysis of affected systems should be performed for plants and plant systems that are identified by the short-term scoping analyses as having the potential for significant system function degradation from small-diameter-line breaks.

6.1.3 Long-Term Implementation Action - If the detailed analysis indicates that significant degradation of the system function results from small-diameter-line break, remedial measures should be taken to preclude small-diameter-line breaks in the affected systems.

6.2 Lines in PWR secondary systems where incidence of SCC has been observed: Current testing, evaluation, and experience indicate limited safety implications in those PWR secondary systems where SCC has occurred.

6.2.1 Short-Term Implementation Action - To the extent practicable, use more SCC-resistant materials on affected PWR secondary systems in plants under construction. In operating plants, replace leaking pipe or pipe where significant flaw indications are found with SCC-resistant materials.

6.2.2 Long-Term Implementation Action - Generic ISI and leak detection requirements should be reevaluated according to the following: For all plants, PWR secondary systems should be ranked according to safety significance and margin against break from SCC. (a) Implement improved and focused inspections for lines with high safety and break potential ranking. (b) For lines with high safety significance and low break potential,

implement improved leak detection where necessary (i.e., outside containment) and reduced ISI. (c) For lines with low safety significance and break potential, inspection requirements should be minimal.

6.3 Thermal fatigue cracking in PWR feedwater lines - Significant thermal fatigue cracking in PWR feedwater lines is unacceptable. Current action taken to deal with feedwater line cracking is satisfactory.

6.3.1 Short-Term Implementation Action - Plants that have experienced cracking should have augmented inspections. If long-term remedial measures are taken and prove to be effective, inspection requirements can be reduced to ASME, Section XI ISI requirements. Operating plants that have not experienced cracking and near-term OLs should have augmented inspections. This augmented inspection should continue until long-term remedial measures are taken or until it is confirmed that these systems are essentially immune to cracking. For other plants under construction and for CP applications, the potential for thermal fatigue cracking should be eliminated.

6.3.2 Intermediate Implementation Action - PWR piping systems and operational procedures should be reviewed to ascertain if areas exist in other safety-related piping where conditions occur that could lead to significant cracking from thermal fatigue.

6.3.3 Long-Term Implementation Action - Systems identified to have the potential for thermal fatigue cracking should receive augmented inspections to ensure cracking which may develop will not go undetected. Remedial measures should be defined on a case-by-case basis. If available data for crack growth rates and flaw characterization as well as fracture mechanics analytical techniques all are acceptable to the NRC and the analyses using these data and techniques indicate that break of a postulated through-wall crack will not occur for ASME Service Level D loadings and the predicted in-service crack depth remains less than 50% of the wall thickness, these remedial actions may be deferred, provided appropriate surveillance programs are approved by the NRC.

References

[1] Shao, L. C., Burns, J. J., "Stress Corrosion Cracking Experience in Piping of Light Water Reactor Power Plants," International Journal of Nuclear Engineering and Design, April 1980.

[2] NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, September 1980.

TABLE I
Summary of Pressurized Water Reactor Feedwater Piping Cracks

Plant	Extent of cracking (nozzle vicinity)			Piping component	Comments
	Maximum depth	Location	Lines cracked		
Westinghouse					
D. C. Cook 1, 2	Through wall	Top	8 of 8	Elbow	2 cracks through wall
Beavey Valley	0.400 in.	9 o'clock	3 of 3	Elbow	13 additional fabrication-related indications repaired
Kewaunee	0.050 in.	7 o'clock	2 of 2	Pipe	3-in. diameter auxiliary feed near SG inlet
Point Beach 1, 2	0.047 in.	3 o'clock	2 of 2	Reducer	3-in. diameter auxiliary feed near SG inlet
H. B. Robinson 2	0.750 in.	9 o'clock	3 of 3	Reducer	Shallow cracking in nozzle under thermal sleeve
Salem 1	0.235 in.		4 of 4	Elbow reducer	
San Onofre 1	0.100 in.	Lower half of reducer	3 of 3	Reducer	Multiple-branched cracks, evidence of some fatigue
Surry 1, 2	0.080 in.	2 & 5 o'clock	6 of 6	Reducer	
Ginna	0.107 in.	8:30 o'clock	2 of 2	Elbow	Cracks also at deep machining marks
Zion 1, 2	0.088 in.				
Combustion Engineering					
Millstone 2	0.250 in.	12 o'clock	2 of 2	Pipe	
Palisades	0.170 in.	3 & 9 o'clock	2 of 2	Pipe	Cracks found also at weld in vicinity of horizontal piping

[1]For SI multiply by 2.5 to convert to centimeters.

Note: A blank cell indicates data unknown.

Table II

Pressurized Water Reactor Facilities Reporting Stress Corrosion Pipe Cracks and Leaks

Facility	System	Pipe size[1]
Arkansas 1	Building (containment spray)	10-in. and 8-in. Schedule 10
	Decay heat removal	10-in. Schedule 10
	Spent fuel pool cooling	3-in. to 2-in. reducer
Crystal River 3	Containment spray	8-in. Schedule 40
Ginna	Safety injection	8-in. Schedule 10
H. B. Robinson 2	Boron injection	4-in.
San Onofre 1	Containment spray	6-in. Schedule 10
	Refueling water pump suction	8-in. Schedule 10
Surry 1	Containment spray	10-in. Schedule 40
Surry 2	Containment spray	10-in. Schedule 40
Three Mile Island 1	Spent fuel pool cooling	8-in. Schedule 40
	Borated water storage tank to residual heat removal suction	10-in. Schedule 40

[1]For SI, multiply by 2.5 to convert to centimeters.

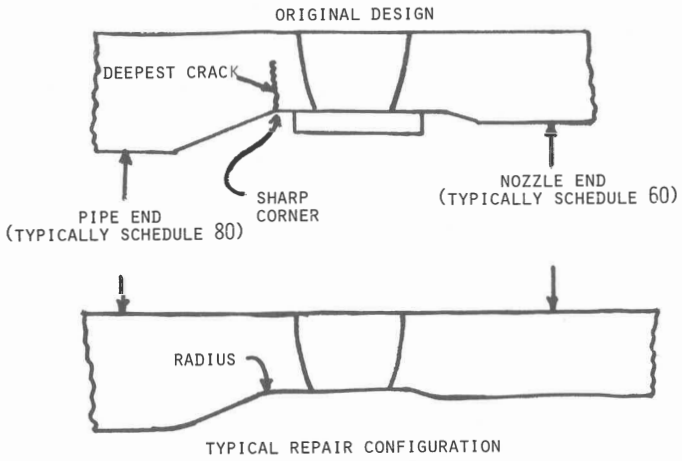

Figure 1. Schematic of Feedwater Piping/Steam Generator Nozzle Vicinity

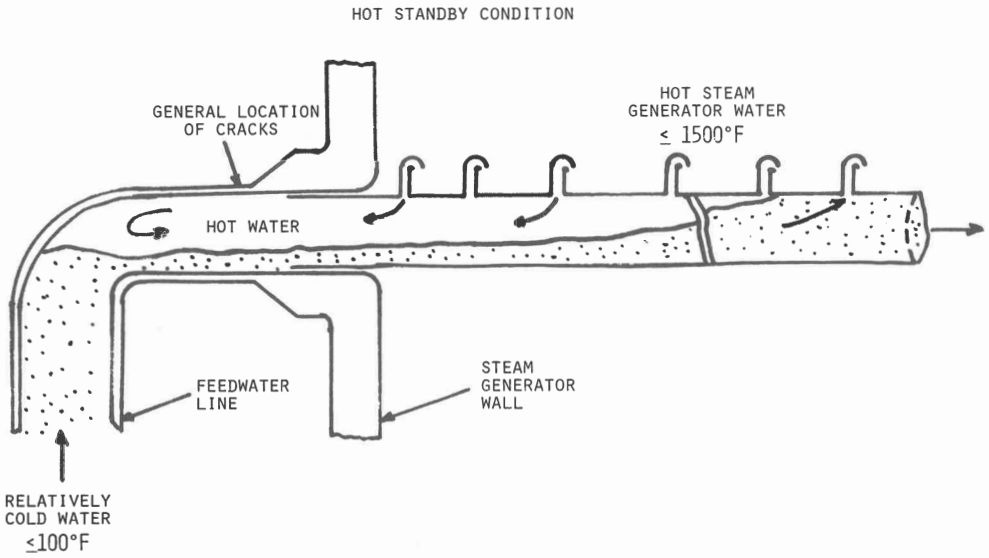

Figure 2. Pressurized Water Reactor Feedwater Line Cracking