Vibration Induced Failures in Nuclear Piping Systems

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The results of earlier work reported in reference [1] show that a very small crack can grow to exceed acceptance standards if the crack is subjected to vibrating loadings with large usage factors. This is particularly true for a surface crack directly influenced by a hot water environment of a Light Water Reactor (LWR). A thick wall pressure vessel was the component considered in the calculations of this earlier work. The work reported in this paper supplements this earlier work. Specifically, this paper reports on the results of a survey that was conducted to determine the extent of cracks that have occurred in piping systems and that are attributable to fatigue.

Preoperational and initial start-up test guidance is given in references [2] and [3] for reactor internals and for other systems important to safety, respectively. In addition, a recently published standard (reference [4]) identifies test program requirements and acceptance criteria for the assessment of piping system vibrations for nuclear plants. The purpose of these tests is to demonstrate that reactor internals and other components important for safety have been designed, fabricated and erected properly to provide the necessary assurances that the facility can be operated in accordance with design requirements and in a manner that will not endanger the health and safety of the public. Corrective actions and modifications to correct deficiencies are to be made if acceptance criteria are not satisfied. Although these tests are also expected to reveal significant vibratory modes, there have been cases where cracks have propagated through pipe walls as a result of fatigue loading conditions.

Most pipe fatigue cracks that have resulted from vibration/fatigue in both BWR's and PWR's have occurred in small lines (less than 4 inches diameter). However, some cracks have also occurred in large feedwater system piping for some PWR plants. Although most of these large pipe cracks occurred prior to 1980, cracks in small pipes have continued to occur. Almost all of the pipe cracks occurred in weld HAZ's and most cracks were detected as a result of leaks.

1. Introduction

A survey of existing documents was conducted to ascertain the extent of pipe crack problems that have been encountered in industry that are attributable to fatigue. The work reported herein is part of a continuing study that the NRC has undertaken regarding crack growth in nuclear components. This paper documents these problems and identifies the reactor plants and the piping system in which the crack(s) occurred. The information was taken from Licensee Event Reports (LER's) from 1969 to October 1982, from Nuclear Regulatory Commission (NRC) Office of Inspection and Enforcement (IE) Bulletins, and from other licensee and vendor reports. References [5] and [6] have been useful as background documents for this effort.

2. Discussion

In earlier work documented in reference [1], it was shown that a very small crack can grow to an unacceptable size (based on the acceptance criteria in Section XI of the ASME Boiler and Pressure Vessel Code) if it is subjected to an oscillatory type loading resulting in a large usage factor. This is particularly true for a surface crack that can be influenced by harsh LWR environments. An 8-inch thick vessel was the component considered in the calculations of this earlier work. As a supplement to the earlier effort, a survey was conducted to determine the extent of crack problems in nuclear piping systems. Table 1* in this paper shows the results of this survey by identifying those plants and piping systems where cracks have occurred due to fatigue loadings. It is apparent from this table that numerous piping components have experienced cracks.

In the LER's, from which most of the data in Table 1 were taken, the inspector's judgment was frequently used in deciding the "cause" of the crack. In the case of the small diameter BWR lines and in some of the PWR large diameter lines, the cause of most cracks was identified as corrosion-assisted fatigue. For these cases, it is impossible to determine from the information given in the LER's which of the causes is the main contributor to crack growth. However, since fatigue was involved, these cases are also recorded in Table 1.

It is evident from the data contained in the LER's and in references [5] and [6] that most fatigue cracks occurred in small diameter (less than 4 inches) pipes. However, fatigue cracks have also occurred in large piping. In addition, most fatigue cracks occurred in welded zones and most cracks were detected as a result of leaks rather than by breaks of the piping. This is true for both BWR and PWR type plants.

*Some of the information contained in this table was taken from reference [5]

Thermal and vibratory service loadings are two types of loadings that can lead to the formation and propagation of fatigue cracks in piping. In the first case, the mixing of hot and cold water as well as continuous changes in thermal profiles across a pipe wall are the mechanisms for initiating and propagating cracks. According to references [5] and [6] thermal fatigue is the primary reason for the cracks in the PWR large diameter piping identified earlier. The second type of loading is related to stresses caused by cyclic forces resulting from pumps or flow induced excitations.

References [2] and [3] are two documents that provide guidance for initial test programs for reactor internals and other systems important to safety, respectively. In addition, a recently published standard, reference [4]*, identifies test program requirements and acceptance criteria for the assessment of piping system vibration for nuclear power plants. Each of these programs is intended to verify the design and to demonstrate that there are no unexpected vibratory responses that could lead to problems during commercial operation. In the case of the latter standard, the basis for the acceptance criteria is consistent with Section III of the ASME Boiler and Pressure Vessel Code; however, these criteria are not consistent with the results of reference [1]. Although some of the plants identified in the table were operational prior to the issuance of references [2] and [3], newer plants have also experienced pipe fatigue cracks.

3. Conclusions

The degree to which the response of a particular piping system is evaluated during initial tests depends on the safety significance of the piping. Although these initial tests are intended to identify potential problems that are to be resolved prior to commercial operation, there have been a number of instances reported in the LER's where additional bracing has been required to reduce the effects of vibration during commercial operation (after through-wall cracks occurred). This implies that either the responses of the piping experiencing the cracks were below acceptance levels during the initial tests, that the piping experiencing the cracks were not evaluated properly, or that the cracks are the result of poor design and/or poor fabrication practices.

*At this time reference [4] has not been endorsed by the Nuclear Regulatory Commission.

Since the LER's do not document specific fatigue information, direct comparisons with the results of reference [1] could not be made. Although the stress levels and whether cracks existed in the pipes initially are both unknown, these results raise more concerns regarding crack acceptance standards and fatigue design requirements.*

To address the concerns raised by the results of this effort as well as the results documented in reference [1], the NRC is sponsoring a research program focusing on the relationship between fatigue design requirements in Section III of the ASME code and the crack acceptance standards in Section XI of the code. This research was initiated in January 1983. Perhaps the results of this research can be reported at the next SMiRT meeting.

*Section III of the ASME Code does not require a fatigue evaluation for Class 2 or Class 3 components.

References

1. W. F. Anderson and E. C. Johns, "Preliminary Analysis of the Effect of Fatigue Loading and Crack Propogation on Crack Acceptance Criteria for Nuclear Power Plant Components," Transactions of the 6th International Conference on Structural Mechanics in Reactor Technology, SMiRT, Volume F, (Paper F6/7), August 17-21, 1981.

2. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Start-up Testing."**

3. Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants."**

4. ANSI/ASME OM-3-1982, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems."

5. L. Frank, W. S. Hazelton, R. A. Hermann, V. S. Noonan, A. Taboada, "Pipe Cracking Experience in Light-Water Reactors", NUREG-0679, August 1980.**

6. PWR Pipe Crack Study Group, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," NUREG-0691, September 1980.**

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TABLE 1

Summary of Pipe Cracking Resulting from Fatigue in LWR Plants

SYSTEM	USA BWR PLANTS	INCIDENCES	USA PWR PLANTS	INCIDENCES
Reactivity Control	Big Rock Point	2	Arkansas 1	1
	Dresden 1	1	Arkansas 2	8
	Dresden 2	2	Calvert Cliffs 1	8
	Dresden 3	3	Calvert Cliffs 2	13
	Humboldt Bay	1	Connecticut Yankee	1
	Peach Bottom 3	1	Crystall River 3	5
	Vermont Yankee	1	Davis-Besse 1	1
			Haddam Neck	2
			Fort Calhoun 1	5
			H. B. Robinson 2	1
			Indian Point 2	7
			Indian Point 3	1
			Kewaunee 1	3
			McGuire 1	1
			Millstone 2	2
			North Anna 1	1
			Palisades 1	3
			Point Beach 2	1
			R. E. Ginna 1	3
			Salem 1	2
			Surry 1	1
			Turkey Point 3	2
			Turkey Point 4	3
			Yankee Rowe	4
			Zion 1	2
			Zion 2	3
Coolant Recirculation	Browns Ferry 1	1	Arkansas 1	1
	Browns Ferry 2	1	Arkansas 2	1
	Peach Bottom 3	1	Calvert Cliffs 2	5
	Vermont Yankee	1	Fort Calhoun 1	1
	Millstone 1	2	Indian Point 1	1
			Millstone 2	2
			Palisades 1	2
			Point Beach 2	1
			Rancho Seco	1
			Salem 1	1
			San Onofre 1	1
			Sequoyah 1	1
			Three Mile Island 1	1
			Turkey Point 4	1
Residual Heat Removal	Browns Ferry 1	2	Arkansas 1	3
	I. Hatch 1	1	Beaver Valley 1	1
	Monticello 1	1	D. C. Cook 1	2
	Peach Bottom 2	3	Indian Point 2	4
	Peach Bottom 3	1	Prairie Island 1	1
	Pilgrim 1	1	Prairie Island 2	1
	Quad Cities 1	1	Salem 1	1
			Three Mile Island 2	1
			Three Mile Island 1	1

TABLE 1 (Cont.)

SYSTEM	USA BWR PLANTS	INCIDENCES	USA PWR PLANTS	INCIDENCES
Reactor Coolant Cleanup	Big Rock Point	2	Calvert Cliffs 1	3
	Browns Ferry 1	1	Calvert Cliffs 2	3
	Monticello	1	Kewaunee 1	1
	Quad Cities 2	2	Maine Yankee	1
			Millstone 2	1
			Trojan 1	2
			Yankee Rowe	1
Emergency Core Cooling	Dresden 2	2	Arkansas 2	3
	FitzPatrick 1	1	Beaver Valley 1	1
	I. Hatch 1	1	Calvert Cliffs 2	1
	Monticello	1	Farley 1	1
	Oyster Creek	2	Fort Calhoun	1
			Millstone 2	3
			Oconee 2	2
			Oconee 3	2
Main Steam Supply	FitzPatrick 1	1	Fort Calhoun	1
			North Anna 1	1
			St. Lucie 1	1
Condensate Feedwater	Big Rock Point	1	Arkansas 1	1
	Browns Ferry 2	1	Beaver Valley*	
	Dresden 2	4	D. C. Cook 1 & 2*	
	Dresden 3	4	Ginna*	
	Millstone 1	1	H. B. Robinson 2*	
	Monticello 1	3	Kewannee*	
	Quad Cities 1	4	Palisades*	
	Quad Cities 2	2	Point Beach 1 & 2*	
			Salem 1*	
			Surry 1 & 2*	
			Three Mile Island 1	2
Other Engineered Safety Features			Turkey Point 3	1
			Cook 2	1
Reactor Core Isolation Cooling	Quad Cities 2	1	Oconee 3	1
Spent Fuel Pool	Pilgrim 1	1	Arkansas 1	2
			Three Mile Island 1	1
Containment Heat Removal			Indian Point 2	1

*More than one pipe in the feedwater system were observed to have cracks.