Materials Reliability Program Fatigue Issues Assessment (MRP-138)

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Materials Reliability Program Fatigue Issues Assessment (MRP-138)

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REPORT SUMMARY

The EPRI Materials Reliability Program (MRP) formed the Fatigue Issue Task Group (ITG) to evaluate fatigue related issues at nuclear power plants. Fatigue failures have occurred at all nuclear power plants and have caused unplanned shutdowns in some cases. To evaluate the recent history of fatigue failures in nuclear plants, this document contains an assessment of fatigue failures that occurred since 1990 along with a discussion of related issues. In addition, the report summarizes the results of work underway at EPRI to develop a comprehensive fatigue degradation matrix for BWRs and PWRs.

Background

In 1991, work was started to develop the EPRI Fatigue Management Handbook (EPRI report TR-104534, Volumes 1-4) to address key issues related to fatigue damage in operating nuclear power plant piping systems and components. The EPRI Fatigue Management Handbook investigated vibrational fatigue failures in small-bore piping and thermal fatigue failures due to thermal stratification. The EPRI handbook was developed to help utility engineers at operating plants understand and act upon fatigue-related issues. Plants have been using the EPRI handbook to evaluate vibration in small-bore piping and thermal fatigue failures due to thermal stratification, but it has been over ten years since the last industry survey on fatigue failures. For this reason, the Fatigue ITG initiated efforts to collect industry data to determine if there are any new trends in fatigue failures that should be addressed. In related work, EPRI is performing work under the Industry Materials Initiative to determine if there may be other longer-term fatigue degradation issues of potential future concern.

Objectives

- To provide an industry assessment of fatigue failures and trending information to determine the main causes of actual plant fatigue failures
- To summarize related work that is attempting to identify where fatigue failures may occur in the future in order to determine the need for additional research related to fatigue.

Approach

This project consisted of three major tasks. First, the project team conducted an industry survey to collect data from plant engineers dealing with fatigue issues. The team sent this survey to each BWR and PWR site to obtain information on fatigue failures that might have occurred at each plant. Second, the team reviewed Institute of Nuclear Power Operations (INPO) plant experience databases to determine the extent of reported fatigue failures occurring from 1990 to July 2004. The results of the surveys and database reviews were combined into a single database and reviewed for trends. Third, the team analyzed other recent related fatigue experience with relief valve chattering. In addition, the team reviewed and summarized the ongoing materials degradation matrix being developed by EPRI.

Results
This report provides an assessment of the fatigue failures that have occurred in nuclear power plants since 1990. It provides trend information on the types of causes that are attributed to fatigue failures. In addition, the report summarizes areas of future potential fatigue-related component degradation. The results of this assessment can be used to determine areas where development of processes to minimize or address the main causes of fatigue failures will maximize plant safety and availability.

EPRI Perspective
Understanding the trends in causes of fatigue failures will assist plant personnel in maximizing plant safety and availability. In addition, it will allow industry to focus efforts on minimizing fatigue failures due to the main causes identified in this report.

Keywords
Fatigue
High cycle fatigue
Vibration
Relief valve chattering
Material degradation

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1
INTRODUCTION

In 1991, work was started to develop the EPRI Fatigue Management Handbook [1] to address key issues related to fatigue damage in operating nuclear power plant piping systems and components. The EPRI Fatigue Management Handbook (herein referred to as the EPRI handbook) investigated vibrational fatigue failures in small-bore piping and thermal fatigue failures due to thermal stratification. The EPRI handbook was developed to help utility engineers at operating plants understand and act upon fatigue-related issues. Plants have been using the EPRI handbook to evaluate vibration in small-bore piping and thermal fatigue failures due to thermal stratification.

In 2000, the Fatigue ITG initiated an effort to collect and evaluate thermal fatigue failure data for PWR normally stagnant non-isolable piping attached to reactor coolant systems. This information was collected to support research related to thermal fatigue failures only [2]. Since it has been over ten years since the last general industry survey on fatigue failures, another effort was undertaken to collect industry data to determine if there are any new trends, especially related to vibrational fatigue failures, that should be addressed. This document contains the results of the industry survey and a review of industry databases on fatigue failures that have occurred from 1990 through July 2004.

An industry survey was developed and sent to over sixty nuclear sites (some sites have more than one plant). In addition, the Institute of Nuclear Power Operations (INPO) databases were reviewed and Event Reports, Operating Experiences, and Licensee Event Reports that pertained to fatigue failures were identified and used to evaluate fatigue problems.

In addition to the industry survey and database review for vibrational fatigue failures, another survey was initiated to determine the extent of failures related to relief valve chattering at nuclear power plants. Ongoing EPRI efforts to develop a comprehensive materials degradation matrix have identified that fatigue is one of the issues that may become more significant as plants age. Thus, the current information from this effort is summarized in this report.

2 DATA COLLECTION AND SCOPE

An industry survey was conducted to evaluate the fatigue problems that have occurred at both BWR and PWR nuclear power plants, primarily due to vibration. Information from this survey included

- Description of the problem(s)
- Affected system
- Potential impact on nuclear safety
- Identification of the source of the fatigue driving mechanism
- What has been done to resolve the issue

In addition to the industry survey, EPRI performed a search of vibrational fatigue failures in the INPO databases that contained plant operating experiences (OE), Licensing Event Report (LERs), and Event Reports. The combination of the industry survey and the INPO database search formed the basis for the fatigue issues assessment.

The scope of the survey was limited to fatigue failures attributed to vibration. A limited review was performed for thermal fatigue failures but the main emphasis was any fatigue failure due to vibration. Thermal fatigue failure experience has been reported in MRP-85 [2].

As part of the fatigue assessment, a separate industry survey was performed to determine the extent of relief valve chattering and its' associated issues.

Industry Survey

An industry survey was sent to twenty-five BWR and forty-one PWR sites. If multiple units exist at a site, the units were treated as one plant for the purposes of reporting survey results. For example, Brunswick Units 1 and 2 were considered one plant even though experience may have been different between the two units. Table 2-1 provides a list of the plants that provided a response to the survey.

INPO Database Search

A search was performed of the INPO databases of plant OEs, LERs, and other Event Reports that identified fatigue failures from 1990 through July 2004. Table 2-1 also lists the plants that reported fatigue failures in the database during this survey period.

Table 2-1
Survey Responses and Other Sources

Plant Name	Utility	Plant Type	NSSS Vendor[1]	EPRI Survey Response	Other Source (INPO, etc.)
ANO 1	Entergy Nuclear South	PWR	B&W	X	X
ANO 2	Entergy Nuclear South	PWR	CE	X	X
Beaver Valley 1, 2	First Energy	PWR	W	X	X
Braidwood 1, 2	Exelon	PWR	W	X	
Browns Ferry 2, 3	Tennessee Valley Authority	BWR	GE		X
Brunswick 1, 2	Progress Energy	BWR	GE	X	X
Byron 1, 2	Exelon	PWR	W	X	X
Callaway	AmerenUE	PWR	W	X	X
Calvert Cliffs 1, 2	Constellation Nuclear	PWR	CE	X	X
Catawba 1, 2	Duke Power Company	PWR	W	X	X
Clinton	Exelon/AmerGen	BWR	GE	X	
Columbia	Energy Northwest	BWR	GE		X
Comanche Peak 1, 2	TXU Electric	PWR	W		X
Cooper	Nebraska Public Power District	BWR	GE	X	
Crystal River 3	Progress Energy	PWR	B&W	X	X
D.C. Cook 1, 2	American Electric Power	PWR	W	X	X
Davis-Besse	First Energy	PWR	B&W	X	X
Diablo Canyon 1, 2	Pacific Gas & Electric	PWR	W	X	X
Dresden 2, 3	Exelon	BWR	GE	X	X
Duane Arnold	Nuclear Management Co, LLC	BWR	GE		X
Farley 1, 2	Southern Nuclear Operating Company	PWR	W		X

Table 2-1 (continued)
Survey Responses and Other Sources

Plant Name	Utility	Plant Type	NSSS Vendor[1]	EPRI Survey Response	Other Source (INPO, etc.)
Fermi 2	Detroit Edison	BWR	GE		X
Fitzpatrick	Entergy Nuclear Northeast	BWR	GE	X	X
Ft. Calhoun	Omaha Public Power District	PWR	CE		X
Ginna	Rochester Gas & Electric	PWR	W		X
Grand Gulf	Entergy Nuclear South	BWR	GE		X
Hatch 1, 2	Southern Nuclear Operating Company	BWR	GE		X
Hope Creek	Public Service Enterprise Group, Inc.	BWR	GE	X	X
Indian Point 2, 3	Entergy Nuclear Northeast	PWR	W		X
Kewaunee	Dominion Generation	PWR	W		
LaSalle	Exelon	BWR	GE	X	X
Limerick	Exelon	BWR	GE		X
McGuire 1, 2	Duke Power Company	PWR	W	X	X
Millstone 2	Dominion Generation	PWR	CE	X	X
Millstone 3	Dominion Generation	PWR	W	X	X
Monticello	Nuclear Management Co, LLC	BWR	GE		X
Nine Mile Point 1, 2	Constellation Nuclear	BWR	GE		
North Anna 1, 2	Dominion Generation	PWR	W	X	X
Oconee 1, 2, 3	Duke Power Company	PWR	B&W	X	X
Oyster Creek	Exelon/AmerGen	BWR	GE		X
Palisades	Nuclear Management Co, LLC	PWR	CE		X
Palo Verde 1, 2, 3	Arizona Public Service	PWR	CE	X	X

Table 2-1 (continued)
Survey Responses and Other Sources

Plant Name	Utility	Plant Type	NSSS Vendor[1]	EPRI Survey Response	Other Source (INPO, etc.)
Peach Bottom 2, 3	Exelon	BWR	GE		X
Perry	First Energy	BWR	GE		X
Pilgrim	Entergy Nuclear Northeast	BWR	GE		X
Point Beach 1, 2	Nuclear Management Co, LLC	PWR	W	X	X
Prairie Island 1, 2	Nuclear Management Co, LLC	PWR	W		X
Quad Cities 1, 2	Exelon	BWR	GE	X	X
River Bend	Entergy Nuclear South	BWR	GE		X
Robinson 2	Progress Energy	PWR	W	X	X
Salem 1, 2	Public Service Enterprise Group, Inc.	PWR	W		
San Onofre 2, 3	Southern California Edison	PWR	CE		X
Seabrook	Florida Power & Light	PWR	W	X	X
Sequoyah 1, 2	Tennessee Valley Authority	PWR	W		X
Shearon Harris	Progress Energy	PWR	W	X	
South Texas Project 1, 2	STP Nuclear Operating Co.	PWR	W	X	X
St. Lucie 1, 2	Florida Power & Light	PWR	CE		X
Surry 1, 2	Dominion Generation	PWR	W	X	X
Susquehanna	PPL Susquehanna, LLC	BWR	GE	X	X
Three Mile Island 1	Exelon/AmerGen	PWR	B&W		X
Turkey Point 3, 4	Florida Power & Light	PWR	W		X
V. C. Summer	South Carolina Electric & Gas	PWR	W	X	X
Vermont Yankee	Entergy Nuclear Northeast	BWR	GE		X

Table 2-1 (continued)
Survey Responses and Other Sources

Plant Name	Utility	Plant Type	NSSS Vendor[1]	EPRI Survey Response	Other Source (INPO, etc.)
Vogtle 1, 2	Southern Nuclear Operating Company	PWR	W		X
Waterford	Entergy Nuclear South	PWR	CE		
Watts Bar 1, 2	Tennessee Valley Authority	PWR	W		X
Wolf Creek	Wolf Creek Nuclear Operating Co.	PWR	W	X	X

[1] B&W = Babcock & Wilcox
CE = Combustion Engineering
GE = General Electric
W = Westinghouse

3 FATIGUE SURVEY RESULTS

The INPO databases were reviewed for instances of fatigue related failures from 1990 through July 2004. In addition, the industry survey provided descriptions of fatigue related failures. Based on the survey responses and database searches, 168 fatigue related failures were identified and were sorted into the following categories:

- Aging
- Design Configuration
- Construction
- Maintenance
- System/Plant Operation
- Other

An additional subset of system/plant operation has also been identified as extended power uprate (EPU), which occurs in BWRs only.

Figure 3-1 shows a plot of the failures versus time for all the nuclear plants and Figures 3-2 and 3-3 shows the number of fatigue failures by type of plant, BWRs and PWRs, respectively.

Fatigue failures, in particular, those due to vibration are typically high cycle fatigue failures that occur in a short time. During initial plant startup, failures due to vibration can occur during the initial startup testing or soon thereafter, since systems are new and may experience vibration and high-cycle fatigue. After the initial plant fatigue failures, there would be a period of plant operation where the number of vibration failures should be fairly low and constant. The theoretical expectation would be at some point in time, as plants approach the end of their design lives, the failure rate will increase again, yielding the classical "bathtub" wear-out curve. A review of Figure 3-1 seems to indicate that the number of fatigue related failures are starting to increase slightly at plants. The fatigue failures also appear to be slightly cyclic in nature; this is probably due to the information that is shared between utilities. Fatigue failures that have occurred at plants are typically reported so that other plants can review similar instances at their plant and avoid the same issue. A review of the vibration fatigue failures shows that there is relatively low repeatability of the same type of fatigue failure at other plants. Specifically, if Plant A had a fatigue failure and reported it to an industry database, other plants were likely to review the failure and perform appropriate actions at their plant to avoid the similar type of fatigue failure.

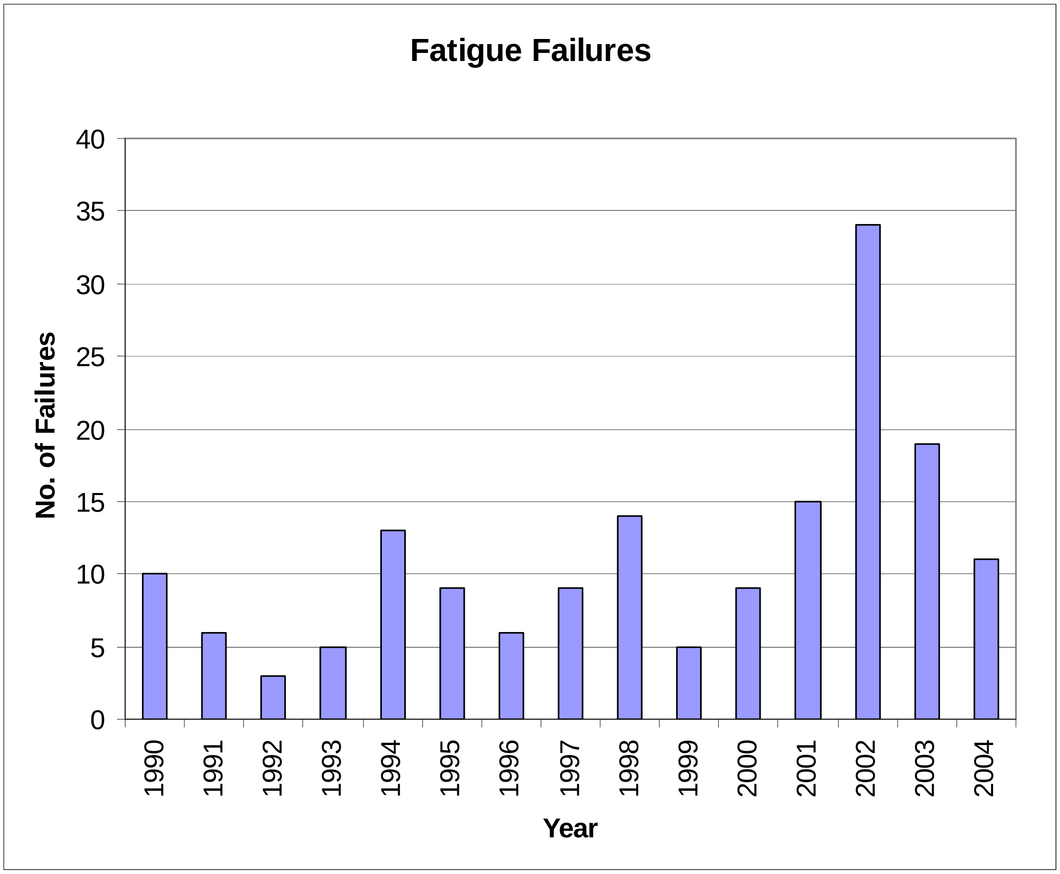

Figure 3-1
Fatigue Failures Sorted by Year

It should be noted that the number of reported fatigue failures for the year 2004 only include the first six months of the year, so the expectation is that the number of failures in 2004 will be greater than those that occurred in 2003. The trend of the fatigue failures appears to be increasing slightly, with a sudden increase in failures in 2002. The fatigue failures were relatively low until the year 2002 where the number of failures more than doubled from previous years. The increase in 2002 is due to small bore pipe fatigue failures which accounted for 13 of the 34 failures. The fatigue failures have been separated into BWR and PWR fatigue failures per year and are shown in Figures 3-2 and 3-3, respectively.

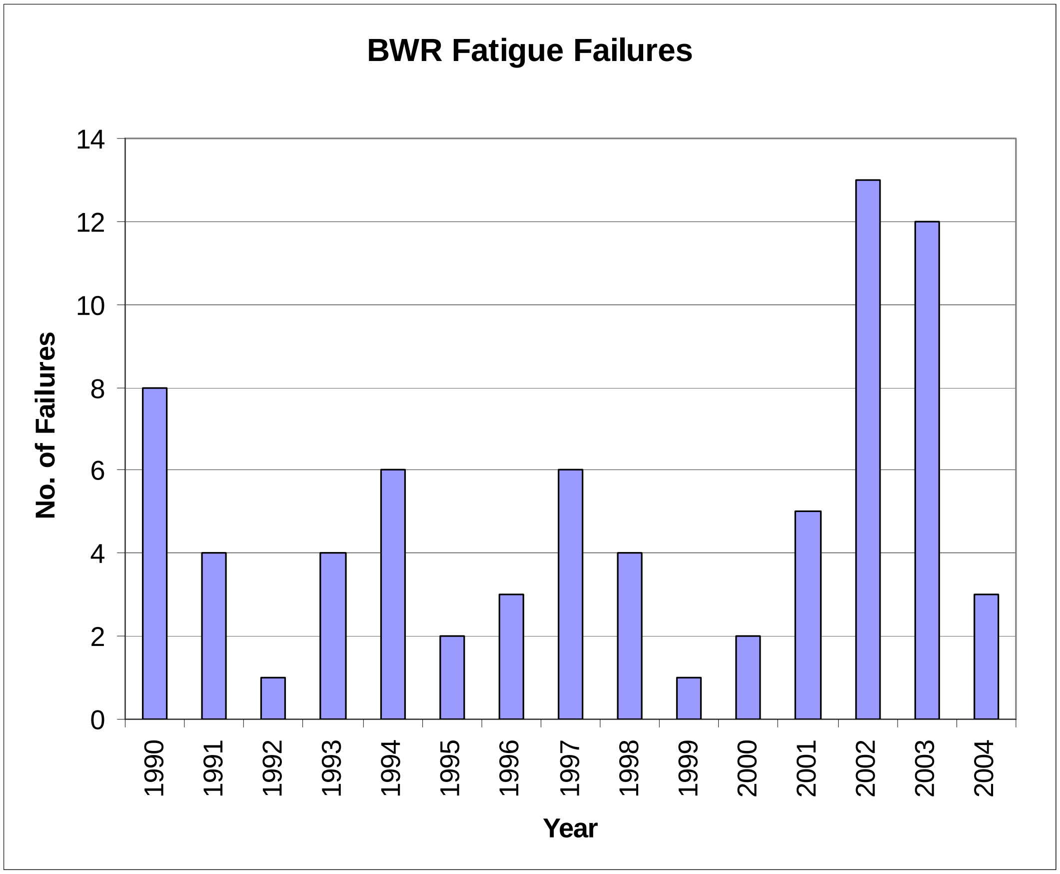

Figure 3-2
BWR Fatigue Failures Sorted by Year

A comparison of Figure 3-2 to Figure 3-3 shows that the trend in fatigue failures is similar between BWRs and PWRs. For BWRs, almost half of the fatigue failures in 2002 are due to small-bore pipe failures and the main cause of the failure was attributed to system/plant operation. The main forcing function that caused the failures in 2002 was flow induced vibration (FIV) loads. A portion of the more recent failures is related to EPU.

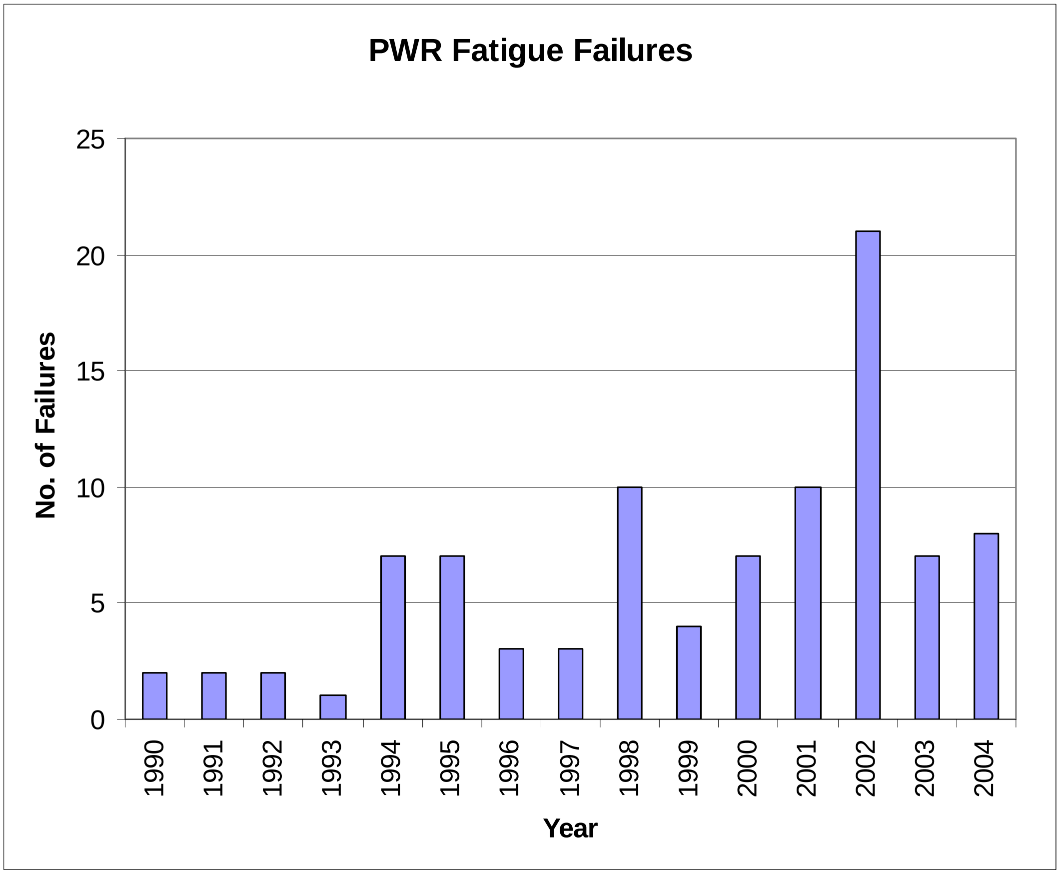

Figure 3-3
PWR Fatigue Failures Sorted by Year

In PWRs, small bore pipe fatigue failures average one to two a year with the exception of 2002 where eight fatigue failures occurred in small bore pipe. The main causes for the failures in 2002 were attributed to design configuration and aging.

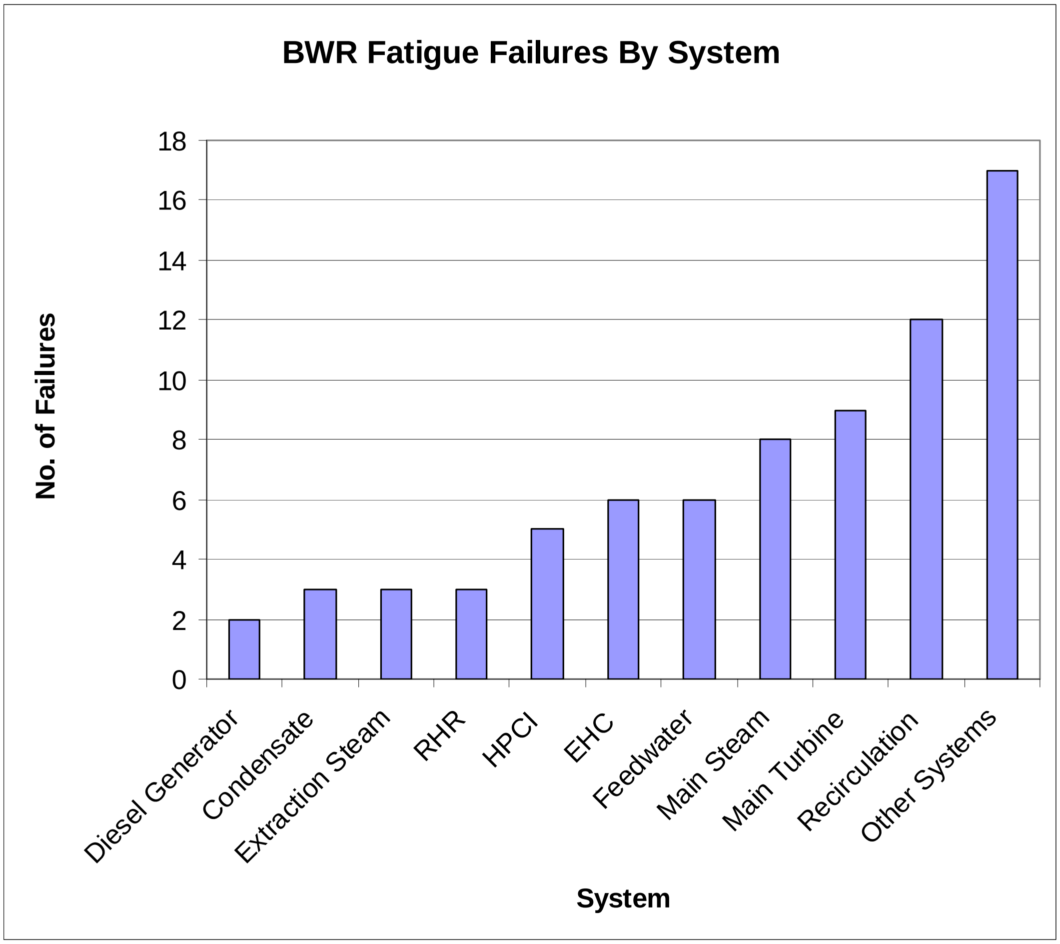

Figure 3-4
BWR Fatigue Failures Sorted by System

Figure 3-4 shows a breakdown by system of the total number of failures occurring in BWRs between 1990 and 2004. In the previous survey [1], the recirculation system had the most failures and remains a high contributor for the current survey period. The "Other Systems" include closed cooling water, HVAC, instrument air, main condenser, main generator, service water, and reactor internal systems.

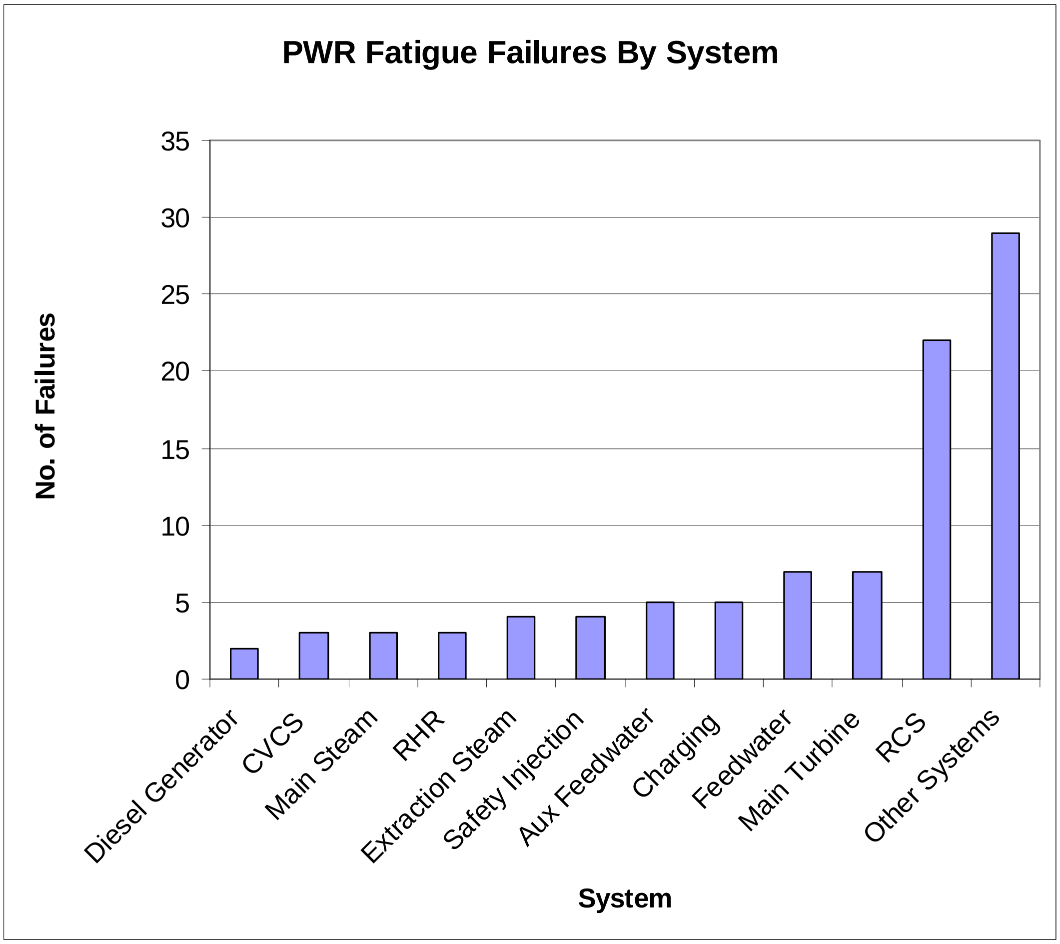

Figure 3-5
PWR Fatigue Failures Sorted by System

Figure 3-5 shows a breakdown by system of the total number of failures occurring in PWRs between 1990 and 2004. In the previous survey [1], the chemical and volume control system (CVCS), consisting of the letdown and charging (or make-up) systems, had the most failures by a large margin. This is not the case anymore; the reactor coolant system (RCS) has had the most fatigue related failures for the current survey period. The previous survey also reported that the next system with the most The "Other Systems" include closed cooling water, HVAC, instrument air, main condenser, main generator, service water, and the steam generator systems.

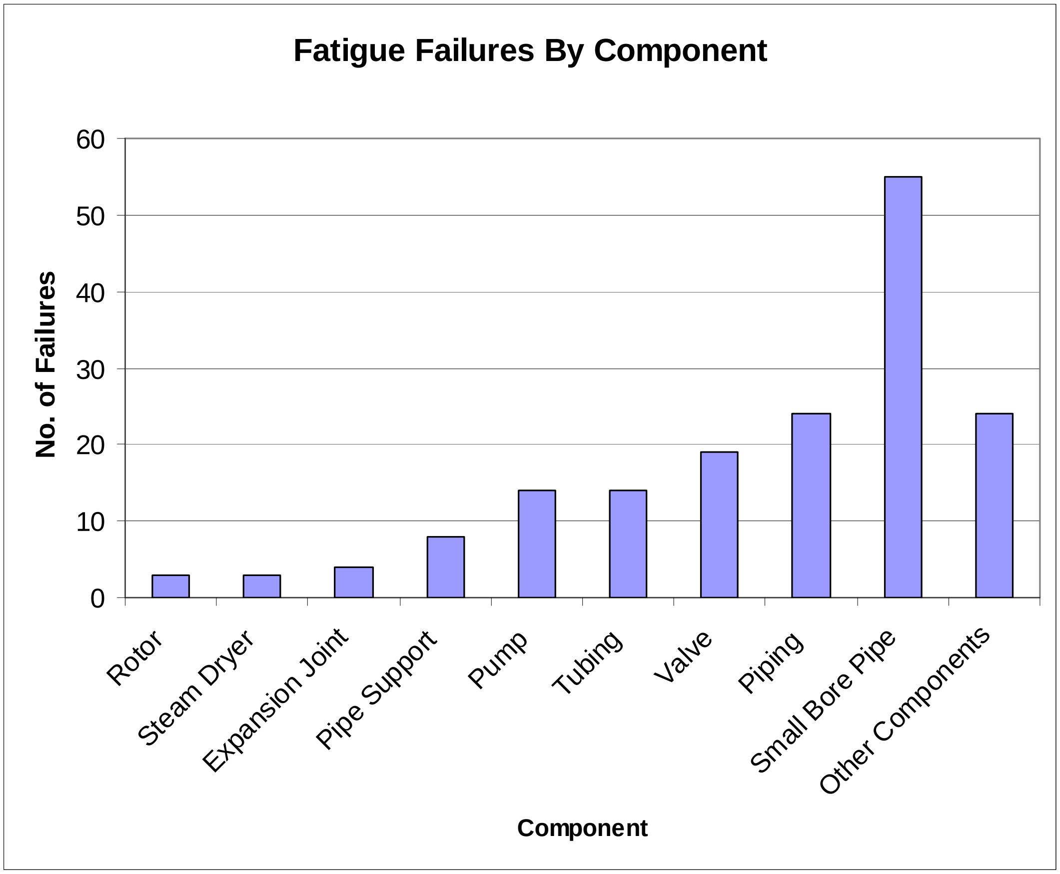

Figure 3-6
Fatigue Failures Sorted by Component

Figure 3-6 shows failures reported in the current survey sorted by component. Similar to the earlier survey, small bore pipe fatigue failures continue to affect plant availability. Some of the valve and piping fatigue failures are welds at valve to pipe connections. Most of the fatigue failures on valves, piping, and small-bore pipe occurred at welds. The "Other Components" include accumulators, bellows, fans, heat exchangers, condenser, and a diesel generator gear. With the exception of the steam dryer which is unique to BWRs, all other component failures occurred in both BWRs and PWRs.

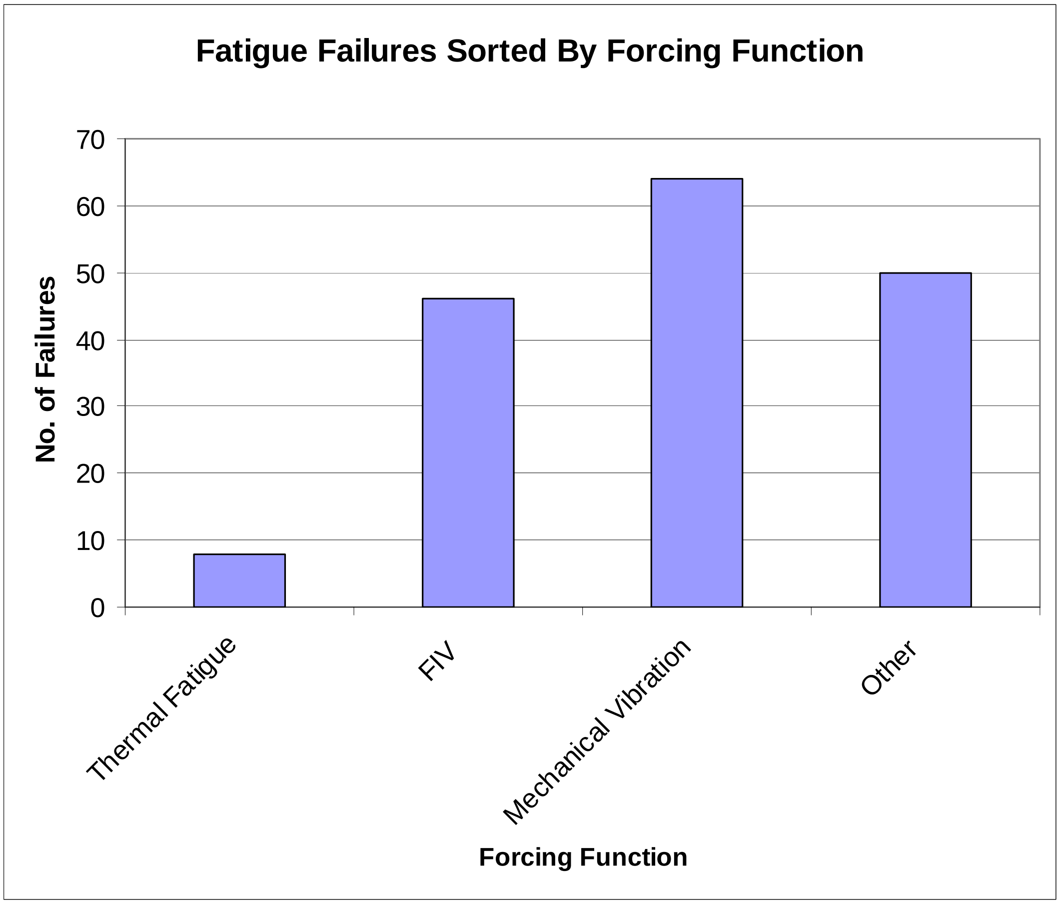

Figure 3-7
Fatigue Failures Sorted by Forcing Function

Figure 3-7 shows the failures sorted by forcing function. The main forcing functions of vibration fatigue failures are mechanical vibration and flow induced vibration loads. The "Other" forcing functions include torsional vibration, water hammer loads, cavitation loads, and forcing functions that otherwise could not be identified. A review of the mechanical vibration events shows that there has been a gradual increase over the years but twelve failures attributed to mechanical vibration occurred in 2002 while only three fatigue failures occurred in 2003. The fatigue failures due to FIV have remained fairly constant over the survey period at around two to three per year.

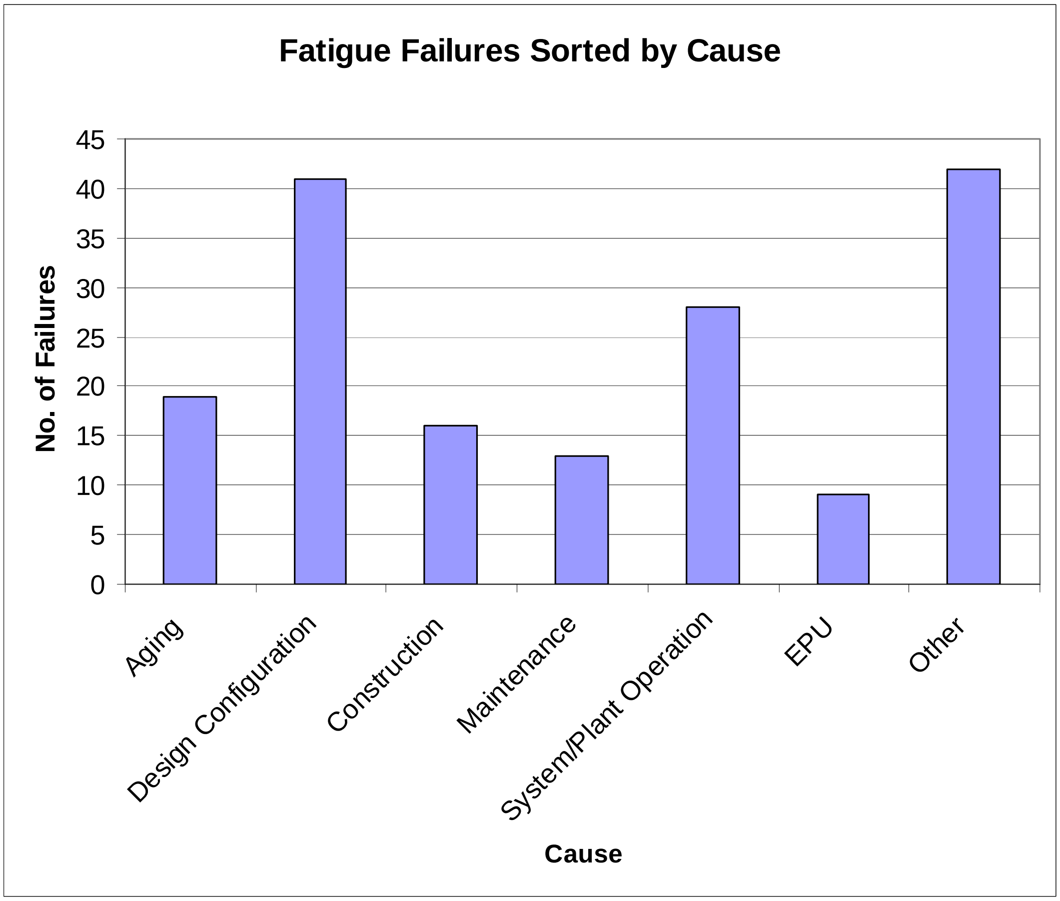

Figure 3-8
Causes of Fatigue Failures

The fatigue failures have been sorted by the cause of the failure in Figure 3-8. The main fatigue failure cause is due to the design configuration. This includes the original design of the system or component that did not properly account for steady state vibration loads. Design modifications to a system or component that result in a fatigue failure are also included in this category. While the number of failures due to design configuration has remained relatively constant over the survey period, design modifications have contributed to the fatigue failures in the recent years. Some of the design modifications are due in part to replacement of components that appear to be the same but have slight differences that change the natural frequency of the component or system. EPU, which is specific to BWRs, is shown as a separate category even though it is a subset of system/plant operation. The “Other” category includes equipment design, manufacturing defects, and unknown causes.

4 TYPES OF FATIGUE FAILURES

The fatigue failures can be sorted into six broad categories

- Aging
- Design Configuration
- Construction
- Maintenance
- System/Plant Operation
- Other

For the purpose of this assessment, the cause of the fatigue failures were classified into a single category when in reality there may be more than one failure cause. To avoid double counting the failures, the primary cause was applied to each fatigue failure.

Aging

One of the causes of fatigue failure is aging, where components gradually wear, clearances increase and cause an increase in vibration. The following sections describe some of the typical fatigue failures that have occurred at plants due to aging.

Feedwater Sample Probe Failure

A failure of a feedwater sample probe occurred at a BWR, which led to damage of the feedwater spargers due to a Reactor Pressure Vessel (RPV) foreign material intrusion. During a refueling outage, an inspection of the sparger revealed a feedwater isokinetic sample probe resting in the sparger. The sample probe was removed and the sparger was repaired. The recovered probe was assumed to be the original feedwater sample probe that had been reported missing in the previous outage.

As a result of the sparger damage, plans were made to conduct internal and external inspections during the next refueling outage. An inspection of the recovered probe concluded that it failed due to fatigue and the most likely mode of failure was mechanical high cycle fatigue due to flow-induced vibration. An analysis of the probe design determined that it was susceptible to vortex shedding and that one of its natural frequencies coincided with the vortex shedding frequency.

Sample probes have been found to be susceptible to fatigue failure but do not fail during the initial plant startup.

This event did not result in an unplanned plant outage but is important because the failed sample probes as foreign material caused damage to the feedwater sparger and could have caused damage to other component in the system such as the feedwater pump impellers, pump casing, and shaft of the condensate booster pumps.

Small Pressure Boundary Leak in Drain Valve

A Combustion Engineering PWR was shutdown due to the discovery of a small pressure boundary leak on a 1-inch drain valve located on the High Pressure Safety Injection (HPSI) system. The leak occurred at a socket weld connection on the upstream side of the drain valve and resulted in a small non-isolable leak. The socket weld failure was attributed to high cycle fatigue. The failed socket weld was repaired by replacement of the entire spool piece and all welds on this section of pipe were modified to a 2 to 1 weld overlay to increase the structural capability of the system. Vibration monitoring of this piping has been enhanced to include review of vibration levels during power ascension. Additionally, measured vibration levels will be validated against stress analysis results to ensure that the analytical models reflect the as-installed piping system.

This event is not significant because the leak was quickly identified, isolated, and repaired but it is important because it resulted in an unplanned plant shutdown to perform repairs.

Main Feedwater Pump Casing Drain and Equalizing Line Cracks

A main feedwater pump casing drain line leak was found at a Westinghouse PWR that caused the plant to reduce power from 100% to 50% to facilitate the repair. This line had been in service for 13 years since the last repair so it did not appear that there was a vibration problem at normal operating speeds. However, the subsequent evaluation determined that resonant vibration occurred during startup and shutdown. A review of the pump operating history showed a correspondence between the failure of the pipe and a startup of the pump. Failure of this line had occurred twice before, the last failure had occurred 13 years ago. The cumulative effects of starting up and shutting down the pump, over a 13-year period, appear to have exceeded the fatigue life of the drain line.

This event resulted in a short 50% power curtailment to perform the repairs. In addition, this type of leak has been occurring for years on these pumps.

Design Configuration

Design configuration is a cause of fatigue failure; this type of failure occurs when the design of a component or system has not been adequately evaluated for steady state vibration. The following sections describe some of the typical fatigue failures that are attributed to design configuration.

Heater Drain Pump Sample Line Failure

A leak was discovered in a BWR heater drain pump sample line involving a sample probe on the heater drain system. The configuration consisted of a ¾ inch stainless steel valve with a short section of pipe attached to a half coupling welded to an 18-inch carbon steel pipe. The leak was on the valve side of the pipe at the toe of the bi-metallic weld. The root cause of this event was determined to be that the design of the piping components was inadequate for the operating conditions. A subsequent evaluation determined that previous industry experience was not effectively utilized to identify and prevent this type of problem. Corrective actions included installing a shortened probe and evaluating the cantilevered piping design that could be susceptible to low stress high cycle fatigue.

This event resulted in an unplanned reduction of power on short notice to repair the failed line.

Main Turbine Electro-Hydraulic Control Leak

A manual reactor scram was required due to a main turbine electro-hydraulic control (EHC) piping leak. The EHC leak was determined to be from a cracked piece of EHC tubing on one of the turbine control valves. A modification to the EHC tubing was performed to address industry issues related to pulsations in the EHC control valve supply lines involved. The modification was installed with no corresponding stress analysis and it created a situation where operational vibrations led to the failure of the tubing. Subsequent analysis was performed and all tubing socket welds were replaced with modified fillet welds that reduced the stresses in the socket-welded joints. Monitoring equipment was also installed to record the vibration levels and ensure that vibration levels were acceptable.

This event caused in an unplanned plant shutdown following discovery of the leak and its subsequent repair.

Main Steam Line Support Failure

A main steam pipe support failed during normal plant operation at a Westinghouse PWR. A section of a steel support had torn away from the remaining structure at the location of the welded bracket to a Wire Energy Absorbing Rope (WEAR) restraint. The 10-inch by 10-inch structural tubing steel support was installed on the main steam system during the previous outage to dampen the main steam line vibration. A metallurgical analysis concluded that the failure of the support was caused by fatigue. The supports that had been installed in this location had failed in the past and a design change was implemented to provide a more robust structure at the location of the failed support.

This event did not cause the plant to shutdown but the loads on this support were finally attributed to flow induced vibration in the piping system due to the configuration of the main steam header located in the turbine building. Numerous root cause evaluations, extensive vibration monitoring, and analyses were performed to resolve the support failure of the main steam line.

Extraction Steam Expansion Joint Failures

Debris from the failure of an extraction steam expansion joint at a Combustion Engineering PWR caused condenser tube leaks. The expansion joint debris consisted mainly of bellows material and external bellows shield collar rings. A large piece of expansion joint debris was discovered in the vicinity of the leaking condenser tube. The two 18-inch expansion joints and their associated piping failed as a result of vibration induced high cycle fatigue. The expansion joint design included the use of Type 312 stainless steel for the bellows and the use of single ply bellows and also lacked tie rods or hinges. In addition, the preventive maintenance tasks to perform visual inspections of the expansion bellows were not sufficient to detect these failures. Corrective actions included the installation of a new expansion joint design that used Type 316 stainless steel and multiple-ply bellows with natural frequencies that are different than the excitation frequencies. The condenser was made stiffer through the addition of pipe guides.

This event did not cause any safety issues but resulted in an unplanned shutdown for one week to repair the damaged expansion joint.

Construction

Construction is a cause of fatigue failure; this type of failure occurs when the construction of the component or system has a defect (e.g., weld defect). The following sections describe some of the typical fatigue failures that are attributed to construction.

Main Generator Hydrogen Supply Line Leak

A leak was discovered in a main generator hydrogen supply line weld of a Westinghouse PWR plant. The leak was found on a 2½ -inch nominal pipe size carbon steel socket weld. This was the final field tie-in weld that was made during installation of the new main generator during the previous outage. The stress initiation appears to be due to a combination of normal background turbine vibration, the design of the socket weld, and a weld defect in the root such as lack of fusion. A pipe clamp and a "leak seal" box were installed to temporarily stop the leak. In addition, pipe supports were added for additional stability. The piping will be returned to original design during the upcoming outage.

This event did not cause the plant to shutdown but the temporary repair will need to be replaced with a permanent fix in the upcoming outage.

Vent Line Socket Weld Failure

A ¾-inch vent line socket weld failure was found on the Reactor Coolant System (RCS) of a Westinghouse PWR. The apparent cause of the crack is a construction groove, evident at the toe of the weld, which induced a stress riser. After several years of operation, the stress riser initiated a fatigue crack; the crack propagated inward, and eventually resulted in a through-wall leak. Corrective actions included replacement of the weld and vibration measurements to confirm that the stresses in the vent line are within allowable limits.

This event did not cause a plant shutdown but it resulted in primary system leakage of borated water.

Reactor Recirculation Sensing Line Socket Weld Failure

The Reactor Recirculation (RR) sensing line socket weld failure at a BWR caused an increase in unidentified leakage, which caused the plant to shutdown. The root cause was attributed to high cycle fatigue crack propagation resulting from severe flow induced vibration by reactor recirculation pump operation at or near the sensing line natural frequencies, coupled with mechanical and welding induced residual pipe stresses. Corrective actions include exclusion zones to minimize recirculation pump operation at or near the sensing line natural frequencies, minimizing residual stresses by enhanced fit-up procedures, and eliminating the mechanically induced stresses by proper support configurations in the repaired sensing line. The weld was also repaired using the 2-to-1 weld leg configuration to further minimize stresses at the socket weld. Vibration monitoring was performed to determine the response of the installed configuration.

This event resulted in an unplanned plant shutdown to repair the failed weld.

Maintenance

Maintenance is another cause of fatigue failure; this type of failure occurs when a system or component fails after refurbishment or testing. The following sections describe some of the typical fatigue failures that are attributed to maintenance.

High Vibration Levels on Reactor Feed Pump

High vibration levels were experienced on a Reactor Feed Pump Turbine (RFPT) following a maintenance activity to balance the rotor. During startup of the pump the turbine experienced vibration levels higher than pre-balance activity levels. The pump was removed from service to address and correct the high vibration levels. The causes of the high vibration were attributed to an incorrect assumption in the vibration data collection setup (wrong detector polarity), which resulted in the balance weights being placed in an improper location to reduce turbine vibration. One of the two installed vibration detectors used to obtain data to calculate the balance shot was manufactured with a polarity 180 degrees from the expected polarity. Corrective actions included the development of a list of critical test parameters that are independently reviewed for the performance of vibration data collection and analysis for balancing the RFPT.

This event did not cause the plant to shutdown but it required equipment to be removed from service.

Diesel Generator Tripped on High Turbocharger Vibration

A diesel generator tripped on high vibration while being loaded during the monthly operability surveillance test at a Westinghouse PWR. The high vibration occurred at the right bank turbocharger. High engine vibration is a non-emergency trip that is bypassed on an emergency start. The cause of the high turbocharger vibration was attributed to an improperly installed turbine oil seal retainer ring that allowed oil to leak into the turbine side of the rotor assembly. Over time, carbon began to build-up on the rotor assembly grooves, which provides the exhaust gas to compressor air torturous path for the labyrinth oil seal. This carbon build-up began to rub thereby causing the high casing vibration. The oil seal retainer ring is removed every time a turbocharger is refurbished in order to replace the turbine side oil seal. Corrective actions include revision of the maintenance procedures to ensure that the holes in the oil seal retainer ring are aligned with the holes in the intermediate case when installed and revising the diesel generator instruction manual to include information and a part number for the turbocharger oil seal retainer ring.

This event did not cause the plant to shutdown but it did involve an equipment failure.

System/Plant Operation

System and/or plant operation is a cause of fatigue failure; this type of failure occurs when a component or system is operated in a manner such that a fatigue failure occurs. System operation includes low flow conditions that may cause cavitation. For BWRs, implementation of Extended Power Uprate (EPU) has caused fatigue failures and is included as a subset of system/plant operation. The following sections describe some of the typical fatigue failures that are attributed to design configuration.

Component Cooling Heat Exchanger Tube Failures

Component Cooling (CC) heat exchanger tubes failed due to excessive flow at a Combustion Engineering PWR. The control room received an indication that the component cooling head tank level was dropping. Six failed tubes in the CC heat exchanger were observed to be the cause of the high component cooling leak rate. Two CC pumps were operated through a single CC heat exchanger. This high flow was sufficient to overexcite the tube bundle into resonance at tubes along a baffle cut, which had very long unsupported spans. These tubes failed due to fatigue, with cracking initiating at pre-existing stress risers. The corrective action included examination and testing of the remaining susceptible tubes and changes to the operation procedure to ensure a CC flow path through both CC heat exchangers during normal operation.

This event did not result in a plant shutdown.

RHR Small Bore Pipe Failures

A BWR experienced small-bore pipe failures on the RHR system due to high cycle fatigue. During an outage with water level at the reactor vessel upper flange, two small bore pipe failures occurred on one of the RHR system loops. One failure occurred at the RHR heat exchanger conductivity cell isolation valve and the second failure occurred in the pressure-locking bypass line for the test return to suppression pool valve. Causes of the piping failures were system operation that was not compatible with the design and a design analysis deficiency. As preparations were underway on the refuel floor to remove the moisture separator, a request was made to reduce the shutdown cooling flow to improve visibility and facilitate efforts to maneuver an underwater camera to help removal of the steam separator. To reduce shutdown cooling flow the RHR heat exchanger outlet valve was throttled. This mode of operation caused the flow velocity to increase at the valve outlet creating cavitation across the valve. Attempting to operate the heat exchanger outlet isolation valve at less than 15% open is not compatible with the valve design. This condition caused excessive vibration that led to the small bore piping failure near this valve.

The second small-bore pipe failure occurred at the pressure locking piping. The design of the pressure locking modification did not account for the pump vane passing frequency so the pressure locking piping modification had a natural frequency that was close to the pump vane passing frequency. The small-bore pipe configuration consisted of an unsupported cantilever with double isolation valves. Corrective actions included a redesign of the pressure locking modification.

This event did not result in a plant shutdown but due to equipment problems resulted in a reduction of water inventory.

EPU Fatigue Failures

Implementation of EPU at BWRs has resulted in vibration fatigue failures that are mainly attributed to FIV or acoustic loads.

Steam Dryer Failures

A BWR was shutdown due to anomalies with reactor pressure and reactor vessel water level, changes in steam flow, and increased moisture carryover. The station was shutdown to determine the cause of these anomalies and a concern that loose parts may have exited the reactor vessel. Upon disassembly of the reactor a section of the steam dryer outer bank hood cover plate was found to be missing. The damaged steam dryer cover plate had broken into several pieces. One section was found on top of the steam separator and was retrieved. No damage to the steam separator had occurred. Another section was separated, but still attached to the steam dryer. The third section was found lodged in the main steam line venturi nozzle. An additional piece was found past the venturi and several small pieces were found upstream of the turbine stop valve strainer. The root cause of the steam dryer failure was lack of industry experience and knowledge of flow induced vibration steam dryer failures. The failure of the cover plate was attributed to high cycle fatigue, as a result of conditions established by the implementation of extended power uprate. Specifically, the higher steam flow rate increased vortex effects in the area adjacent to the cover plate. This vortex set up a standing wave of 180 Hz, which matched the natural frequency of the cover plate. Repairs were made and the cover plate was strengthened.

This event resulted in an unplanned shutdown, reactor disassembly, loose parts in the reactor vessel and main steam system, and over ten days of lost power generation.

Excessive Valve Actuator Wear

During an outage a severed pilot vent line was found on an Electromatic Relief Valve (ERV). As repair work was performed on the ERV, maintenance workers identified damage to the ERV solenoid actuator. The broken line was removed and sent to a test lab for failure analysis. The lab identified mechanical indentations in the pipe indicative of a chipping hammer, which is routinely used to remove slag during welding activities. The lab further identified weld voids within the same area at the site of the failure.

During removal of the solenoid actuator cover, the cover was found to have broken into several pieces as a result of tack weld and seam weld failures. Initial inspection of the interior of the solenoid actuator revealed two component failures, the protrusion of a plunger spring through a brass bushing on the plunger guide, and a missing actuating arm and spring for one limit switch. Bench testing indicated that the solenoid would not have operated upon demand. System modeling and vibration analyses were utilized to determine the potential failure modes. In addition to the analytical results, the ERV was dynamically tested on a shake table to confirm the proposed design modifications. Shake table testing showed that a local mode of the plunger caused the excessive bushing wear. Modifications to the actuator were installed on the actuator and tested on the shake table to confirm acceptability of the modifications. The excessive wear was attributed to increased vibration following implementation of extended power uprate.

5
RELIEF VALVE CHATTERING SURVEY

Piping fatigue can result from many sources including equipment vibration, flow induced vibration, etc. One potential source of excessive fatigue loading on piping is liquid relief discharges that result in valve "chatter". Relief valves are more susceptible to chatter (high frequency oscillation of the valve disk) when the valves lift under liquid conditions. The potential for chatter is higher for relatively colder water and for long inlet piping. During a chattering event, the relief valve inlet piping can be subjected to a large number of cycles with high-pressure oscillations. These vibrations can also affect welds of nearby attached piping.

A survey for relief valve chattering was conducted. Responses indicated that only two specific plants could identify instances where the problem with a relief valve was specifically attributed to chatter.

One PWR plant that responded to the survey experienced relief valve chattering in the region downstream from the Chemical and Volume Control System pumps. Following a plant trip, loss of pressurizer level caused the two normally-standby pumps to start, creating a pressure surge. Three relief valves lifted. Since the water is cold, and the lines to the relief valves were relatively long, pressure was lost beneath the valve seats when the valves opened. This resulted in opening and closing of the valves (chattering). All valves were damaged due to the internal hydraulic loadings associated with the valve chatter phenomenon.

The other response was from a BWR plant. At this plant, three occurrences of multiple valve lifts were reported, with one occurrence each for the feedwater system, the shutdown cooling (RHR) system, and the reactor water cleanup system. In two of the three cases, the occurrence of multiple lifts resulted in not being able to reseat the valve fully such that it was leak tight.

The response from this survey does not indicate that this is a widespread industry issue for US plants.

In France, there have been several incidences similar to that at the US PWR plant. In this case, the French regulatory authority has required that specific actions be taken by plant operators to assess previous experience and to modify certain piping welds if necessary. This program is ongoing.

6 DISCUSSION OF VIBRATION FATIGUE FAILURES

A review of the fatigue failures gathered in this survey show that the general trend appears to be increasing slightly. Some of this can be attributed to vibration shakeout of systems or parts of systems that are not used for normal plant operation. Bypass loops or backup systems that are typically only used to perform a test accumulate cycles at a slower rate than systems that are used constantly. Systems that are not used frequently and are only tested periodically are also starting to experience fatigue failures. It takes more time to accumulate the number of cycles required to reach fatigue failure for these system components that do not have high normal use.

Plant operation has changed over the years as outage time is reduced and plants perform more on-line testing. Modifications to system operation occur as plants minimize their outages and perform more tests on-line. For example, the pressure locking piping that failed as part of the RHR small bore pipe failures described in the System/Plant Operation section was installed to allow on-line testing. Plant operation required the use of double isolation valves to allow on-line testing but the modification was installed on an unsupported cantilever pipe that eventually failed.

Implementation of EPU at BWRs has caused some additional fatigue failures that were not expected. The development of the acoustic forcing function on the steam dryer is a complex issue that needs to be addressed to determine appropriate modifications to the steam dryer. Flow induced vibration issues, while not increasing, still exist at plants and plant personnel need better tools to be able to evaluate FIV loads on components.

As plant personnel retire, some of the plant history may be lost. This loss of historical knowledge can affect those fatigue failures that are due to system or plant operation. Lessons learned during initial plant startup and normal operation may be lost and failures that were solved in the past may occur again. Specifically the reason why something is operated in a particular manner may not be documented so when a modification is implemented, failures can occur due to the modification.

To help plant personnel, the EPRI Fatigue Management Handbook could be updated to include techniques to solve vibration fatigue failures. The EPRI handbook provides guidance and screening criteria for susceptible small bore pipe configurations but does not specifically address how to solve a vibration fatigue failure. It could be enhanced to include evaluation techniques to solve vibration fatigue failures.

7 OTHER POTENTIAL FATIGUE ISSUES

As part of the current Industry Materials Initiative, EPRI has undertaken an effort to develop a Materials Degradation Matrix. Industry experts have been solicited to identify areas where failures may potentially surface in the future. In this section, the work in the area of fatigue is summarized. Since the materials degradation matrix has not been finalized, the following is a snapshot in time. It is expected that future efforts, by the Fatigue ITG and others, will perform additional evaluations to focus on the higher potential issues.

Equipment

The Degradation Matrix recognizes that fatigue is an aging degradation mechanism that can affect a number of major components throughout the reactor coolant pressure boundary of both PWRs and BWRs. Major components potentially affected by fatigue include:

1. Reactor coolant system (RCS) piping, fittings and valves and RCS-attached branch line piping/fittings/valves
2. Reactor pressure vessel (RPV)
3. Pressurizer
4. Steam generator shell, nozzles, divider plate, tubesheet, tubes, and internal structures
5. RPV internals components
6. PWR reactor coolant pumps and BWR recirculation pumps

Materials

A large variety of materials are used in the above components, including:

1. RCS Piping and fittings – carbon steel (CS), low-alloy steel (LAS), stainless steel (SS), and cast austenitic stainless steel (CASS)
2. Reactor pressure vessels (applicable to both PWR and BWRs) - low-alloy steel (LAS), wrought stainless steel (SS) cladding, wrought nickel-based penetrations and various weld materials depending on the parent material used.
3. Pressurizer – same as reactor pressure vessels
4. Steam Generator shell, tubes, and internals – same as reactor pressure vessels plus (steam generator tubes)
5. RPV Internals (applicable to both PWR and BWRs) – SS, cast austenitic stainless steel (CASS), Inconel, and various weld materials depending on the parent material used.
6. Pumps – SS and CASS for pressure boundary materials; various high alloy steels for bolting and austenitic or martensitic SS for pump shafts and other internal components.

Fatigue Degradation Mechanism and Mitigation Options

Fatigue is the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized micro-structural damage has been accumulated, crack initiation can occur at the most highly affected locations. Subsequent cyclic loading and/or thermal stress can cause crack growth. The relevant fatigue-related degradation mechanisms include:

1. High-cycle fatigue – The most ‘classical’ fatigue-related degradation mechanism is high-cycle (HC) fatigue. High cycle fatigue involves a high number of stress cycles at a relatively low stress amplitude (typically below the material’s yield strength but above the fatigue endurance limit of the material). High cycle fatigue may be:
   a. Mechanical in nature, i.e. vibration or pressure pulsation or due to flow-induced vibration (FIV). FIV can induce HC fatigue in otherwise normally passive components merely through interaction of flow adjacent to the component or within the system, establishing a cyclic stress response in the component. Additionally, power uprates are also of major concern since an increase in flow may change the acoustical characteristics of the system and excite a HC mode where resonant frequency is achieved.
   b. Thermally induced due to mixing of cold and hot fluids where local instabilities of mixing lead to low-amplitude thermal stresses at the component surface exposed to the fluid. Thermal fatigue occurs at mixing tees where the temperature of injected fluid is different than that of the flowing fluid. There are a number of research programs underway in France [3,4] to investigate the phenomena associated with mixing tee fatigue initiated after the Civaux RHR piping leakage [2]. Bypass at thermal sleeves can cause a similar effect. Thermal striping at the interface of hot and cold fluids that are stratified can lead to high-cycle fatigue.
   c. Combinations of thermal and high cycle mechanical loads such as might occur on pump shafts in the thermal barrier region.
2. Low-cycle fatigue – Low cycle fatigue is due to relatively high stress range cycling where the number of cycles is less than about 10^4 to 10^5. To induce cracking at this number of cycles, the stress/strain range causes plastic strains that exceed the yield strength of the material, and the cycling causes local plasticity leading to more rapid material fatigue degradation. In reactor coolant system components, the cumulative combined effects of reactor coolant system pressure and temperature changes are considered in the component design analysis. The stress cycling that contributes to low cycle fatigue is generally due to the combined effects of pressure, attached component loadings (e.g., piping moments) and local thermal stresses that result during normal operation. Thermal stratification in piping systems can produce low-cycle fatigue due to cycling between stratified and non-stratified conditions.
3. Thermal fatigue – Thermal fatigue is due to cyclic stresses that result due to changing temperature conditions in a component or in the piping attached to the component. Thermal fatigue may involve a relatively low number of cycles at a higher stress (e.g., plant operational cycles or injection of cold water into a hot nozzle) or due to a high number of cycles at low stress amplitude (e.g. local leakage effects or cyclic stratification).

4. Environmental effects – Environmentally-enhance fatigue concerns the reduction in fatigue life in reactor water environment compared to that in room temperature air environment. Environmental fatigue involves two primary elements; the effects of a reactor water environment on the overall fatigue life of reactor components (as represented by either multiplying the location fatigue usage factor by a 'factor' to account for environment or use of an environment-adjusted fatigue design curve), and the potential accelerated growth of an identified defect due to reactor water environments. With regard to the evaluation of fatigue for component aging management, consideration of the effects of a particular reactor water environment on the overall fatigue life is more relevant. Environmental acceleration of fatigue crack growth is important in dispositioning detected/postulated flaws in a component to permit continued operation.

Fatigue crack initiation and growth resistance is governed by a number of material, structural and environmental factors, such as stress range, temperature, fluid oxygen content, mean stress, loading frequency (strain rate), surface roughness and number of cycles. Cracks typically initiate at local geometric stress concentrations, such as welds, notches, other surface defects, and structural discontinuities. The presence of an oxidizing environment or other deleterious chemical species can accelerate the fatigue crack initiation and propagation process. For example, oxidation can produce pits in the surface of some alloys that act as stress concentrators and potential fatigue crack initiation sites.

Design against fatigue damage is based on fatigue curves in Section III of the ASME Boiler and Pressure Vessel Code. These curves indicate the number of stress cycles of given amplitude of stress intensity that is required to reach a usage factor of 1.0. The fatigue curves are based on test data taken in air at room temperature reduced by a factor of 2 on stress or a factor of 20 on cycles to failure, whichever is most conservative, to account for scatter of data, size effects, roughness, and industrial environment. For carbon and alloy steel materials, the most adverse conditions of mean stress are used to correct the test data prior to applying these factors. The ASME Code includes analytical approaches and criteria for determining usage factors for Class 1 components. For Class 1 Code components, the usage factor must be shown to be less than 1.0 for the component life. However, a fatigue usage factor of unity does not imply crack initiation because of the safety factors applied to the stress amplitude or number of allowed cycles for the Code fatigue curves. At the same time, the Code curves were based on room temperature testing in air, so do not explicitly address environmental effects associated with light water reactor coolant.

The crack growth that follows fatigue crack initiation can be predicted if the crack can be characterized and if the cyclic stress field is known. Procedures for performing crack growth analyses are contained in Section XI of the ASME Code.

Mitigation of fatigue damage for existing components is accomplished by reducing the magnitude of the applied loads or thermal conditions or by reduction of the number of cycles of loading. For thermal transients, reduction in the rate of temperature change for extreme temperature cycles can be effective. However, the normal operating cycles are not generally the source of significant fatigue damage in nuclear plants. The observed fatigue cracking has generally been due to high cycle fatigue as a result of conditions not known at the time of original plant design.

Major areas where fatigue failures and leakage has occurred, or might be expected to occur in the future include the following:

1. RCS Piping - A number of fatigue issues have been identified. The major occurrence of leakage has been due to mechanical vibration-induced cracking of small attached lines (primarily socket welded instrument lines).
   a. Power uprate has contributed to a number of recent incidences. Thermal fatigue has also caused cracking in normal flowing lines where relatively colder water is injected into flowing RCS lines.
   b. Thermal fatigue has also occurred in a number of normally stagnant branch lines attached to flowing RCS lines. The source has been thermal stratification/cycling due to valve in-leakage in up-horizontal running safety injection line configurations and swirl-penetration thermal cycling in down-horizontal drain/excess letdown lines. This is being addressed by the MRP Fatigue ITG, where guidelines are planned in mid-2005. An interim guideline was issued in 2001.
   c. Although no occurrences of leakage have been identified, an issue related to surge line stratification was identified in 1988 [5]. The issue was resolved by analysis; however, the computed usage factors were quite high. Environmental fatigue effects may be significant for these lines.
   d. Other potentially susceptible locations include PWR charging nozzles and BWR RHR tees, where significant thermal transients can occur in some plants and where thermal mixing occurs during normal operation or other plant operational modes.
2. Reactor Pressure Vessels – The effects of fatigue are currently managed by adherence to the plant design basis, where thermal transients were considered in the original plant designs. The notable exceptions are BWR feedwater nozzles and control rod drive nozzles, where the effects of cold water injection caused cracking early in the plant life. Mitigating actions and continued monitoring have been implemented. There may be other areas affected, such as vessel cladding interface, flange-shell connections, connections to reactor internal component supports, etc.
3. Pressurizers – There have been no known fatigue failures in pressurizers. However, recent consideration of cold water insurge to pressurizers has been identified that may be a contributing factor to leakage that has been observed in pressurizer heater sleeve welds. The pressurizer spray nozzle is also affected by some significant thermal transients, especially due to auxiliary pressurizer spray. Pressurizer surge nozzles can be affected by thermal stratification conditions in the surge line.
4. Steam generator components – Steam generator feedwater nozzles have exhibited cracking as a result of thermal stratification and cycling, but high oxygen content of the feedwater for low-power conditions may have also increased environmental effects. Girth weld cracking of the steam generator shells and feedwater nozzle blend radii have also been observed, where both hot/cold water thermal fatigue and an environmental contribution were identified.
5. RPV internals components – The major issue identified has been that due to flow induced vibration of BWR steam dryers. This has led to cracking of the vessel-attached support brackets on several plants. Also, BWR jet pumps and related components have have experienced damage due to vibration.

The area of environmental fatigue is still evolving and is under considerable discussion in the technical community, Code bodies, and regulatory agencies. Laboratory data indicates that the reactor environment leads to less fatigue resistance of materials than is represented by room temperature testing in an air environment. Code safety factors may bound some of this difference. In addition, the effects of flow adjacent to affected components may reduce some of the environmental effects. Work continues to better quantify these effects and to determine how these effects may be factored into Code acceptance criteria for design.

8
SUMMARY AND RECOMMENDATIONS

A review of the fatigue failures gathered in this survey show that the general trend appears to be increasing slightly. Fatigue failures between BWRs and PWRs are relatively similar with both types of plants experiencing a larger than expected increase in vibration fatigue failures in 2002. Fatigue failures dropped back to expected levels in 2003 but with only half of 2004 included in the survey, the fatigue failures in 2004 could easily exceed those in 2003.

The fatigue failures were sorted by system, component, forcing function, and cause of failure. The fatigue failures were sorted into six causes of failure categories

- Aging
- Design Configuration
- Construction
- Maintenance
- System/Plant Operation
- Other

Examples of each failure cause were provided in Section 4.

It is recommended that the EPRI Fatigue Management Handbook be updated to help plant personnel solve vibration fatigue failures. The EPRI handbook provides guidance and screening criteria for susceptible small bore pipe configurations but does not specifically address how to solve vibrational fatigue failures. The EPRI Handbook should be enhanced to include evaluation techniques to solve vibration fatigue failures.

The other issue of significance is that due to extended power uprate in BWR plants. Here, flow rates are being increased significantly, leading to new sources of flow-induced loadings on piping components that have lead to vibration. A similar phenomenon has been seen in PWR main steam and feedwater piping, but the effects are not so pronounced due to the smaller changes. This is another area where additional information would be useful in helping utility engineers solve vibration problems. This could also be included in an update of the EPRI Handbook.

From review of the EPRI work related to development of a degradation matrix for fatigue, it was concluded that the combined effects of adverse loadings and environmental effects might lead to more cracking than has been observed in the past. In addition, the effects of power uprate have increased the occurrences of flow induced vibration failures and related damage to component supports. Thus, research in the following areas is recommended:

1. Develop a better understanding of the relationship between laboratory environmental testing and actual reactor water conditions. The conditions in laboratory testing are significantly different than that observed in actual reactor water flowing conditions. In addition, material conditioning between the extremes of actual cyclic conditions may be beneficial in reducing environmental effects. Although this has been primarily identified as a License Renewal issue, the laboratory effects are real and indicate that the fatigue resistance in a water environment is not as good as previously understood.
2. Investigate high cycle fatigue effects due to hot and cold water mixing. Several incidences of cracking in France have led to Electricitė de France embarking on research programs in this area. Participation in these efforts and determination of applicability to all regions of mixing in US plants should be undertaken.
3. Improve methods for predicting and quantifying flow-induced vibration and acoustic loadings. A number of cases have been identified that have resulted in piping and component wear and failure.
4. Past attention to fatigue issues have related primarily to pressure-retaining components. Additional more detailed evaluations are needed to determine flow-induced fatigue effects and safety/reliability consequences for reactor internals and possibly other reactor components.

9
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