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U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Interim Thermal Fatigue Guideline

PROJECT NUMBER: 689

Dear Mr. Strosnider:

The EPRI Materials Reliability Program (MRP), *Interim Thermal Fatigue Guideline (MRP-24)* is enclosed for your information. This document was recently provided to the industry for assessing thermal fatigue of reactor coolant piping systems.

During the past few years, several domestic and foreign plants experienced thermal fatigue cracking in stagnate-flow piping attached to PWR main reactor coolant systems. In 1998, NEI, the MRP and the NRC discussed a concern that the ASME Code required surface examination would not detect thermal fatigue in small diameter high-pressure safety injection piping (Class 1 piping). The MRP evaluated this concern and formed the Thermal Fatigue Issue Task Group (ITG) to develop a guidance document to assess thermal fatigue in Class 1 piping systems.

In late 1999, the ITG decided to develop an interim guideline since the final guideline would not be available until mid-2002. The interim guidance provides evaluation and inspection recommendations for determining if a potential exists for thermal fatigue in systems with normally stagnate-flow. The scope of the interim guidance is limited to locations that have previously experienced thermal fatigue in domestic or similar foreign plants, but are not currently part of another augmented inspection program. The guidance also provides screening criteria to identify piping lines that are not susceptible to cracking.
The ITG discussed the proposed interim guidance with the NRC staff in late-2000. The enclosed guide was published after considering the NRC staff comments.

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Please direct any questions to Kurt Cozens at (202) 739-8085, koc@nei.org or me.

Sincerely,

Alex Marion

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KOC/maa
Enclosure

c: Mr. Keith Wickman, U. S. Nuclear Regulatory Commission
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EPRI

Interim Thermal Fatigue Management Guideline (MRP-24)

Technical Report

Interim Thermal Fatigue Management Guideline (MRP-24)

1000701

Final Report, January 2001

EPRI Project Manager
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REPORT SUMMARY

The Materials Reliability Project (MRP), formed in January 1999, is an association of utilities focusing on PWR vessel, material, and related issues. Its Thermal Fatigue Issue Task Group, formed in mid-1999, is evaluating the potential effects of thermal fatigue on normally stagnant piping systems attached to reactor coolant system piping as they might be affected by valve in-leakage or turbulence penetration effects. This report, based on research to date, provides interim guidelines for evaluating and inspecting regions where there might be high potential for thermal fatigue cracking that could lead to leakage and forced plant outages.

Background

In 1998, the Nuclear Regulatory Commission expressed concerns that the surface examinations of small diameter (< 4-inch nominal pipe size) high pressure safety injection piping required by ASME Section XI were not adequate and that volumetric examination should be considered. These questions led to formation of the MRP Thermal Fatigue program to provide evaluation and assessment techniques that would identify if additional inspection was required or not.

Objective

To provide a common industry approach that may be used to assess the potential for thermal fatigue cracking in piping systems where through-wall leakage has been observed in other plants in the past. This interim report is meant to provide early feedback to PWR plant operators prior to completion of this MRP project.

Approach

The MRP thermal fatigue project is a multi-tasked effort to provide screening, evaluation, monitoring, inspection, operations, maintenance, and modification guidance to utilities to avoid damage due to thermal fatigue. Based on ongoing results from this program, the project team developed this set of interim guidelines to provide timely feedback to utilities so that there is an awareness of some key locations where thermal fatigue cracking may potentially occur. In addition, the team reported on newly developed non-destructive examination methods that can be used to detect thermal fatigue cracking and crazing.

Results

These interim guidelines provide assessment criteria that will allow utilities to determine if normally stagnant safety injection and drain lines attached to the reactor coolant system might be affected by thermal fatigue. Based on this assessment, specific locations may be identified for inspection or further evaluation. If susceptible locations are identified, the guidance provided in the report may be used to perform effective nondestructive examinations or to implement monitoring to effectively manage thermal fatigue cracking.

EPRI Perspective

These interim guidelines provide utilities with a simple set of evaluation criteria that can assist in identifying some key locations that might be affected by thermal fatigue conditions not known at the time of initial plant design. Use of these guidelines and the associated nondestructive examination techniques can contribute to effectively managing thermal fatigue and assist in avoiding unplanned outages due to thermal fatigue cracking.

Keywords

Fatigue
Thermal Fatigue
Inspection
Nondestructive Examination
Reactor Coolant Piping
Cracking
Leakage

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Others:
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EXECUTIVE SUMMARY

In early 1998, there were discussions between the nuclear industry and the Nuclear Regulatory Commission (NRC) regarding additional volumetric examination of Class 1 high pressure safety injection piping with diameter less than 4-inch nominal pipe size. Based on several instances of thermal fatigue cracking in safety injection lines (at Oconee, Farley and several foreign plants), NRC concluded that surface examination was not sufficient to assure the integrity of this reactor coolant pressure boundary piping throughout the design life. The industry position was that inspection of all safety injection piping was not required and that alternate evaluations and/or monitoring programs could provide adequate assurance that leakage would not occur. In addition, it was questioned if volumetric examination would be fully effective in detecting fatigue cracking in the smaller diameter piping systems.

In March 1999, creation of the Thermal Fatigue Issue Task Group (TF-ITG) was approved by the MRP Executive Group and Senior Representatives. In mid-1999, a preliminary program was described to the NRC consisting of tasks related to screening and analysis, monitoring, modifications and related tasks, culminating in final guidelines for thermal fatigue management in late 2001. Because of NRC concerns that there were no more immediate licensee actions, it was committed that interim guidelines would be available by mid-2000.

This guideline presents interim assessment and examination recommendations for determining if there may be potential thermal fatigue cracking in normally stagnant piping systems attached to Pressurized Water Reactor (PWR) main reactor coolant piping. The objective of this guideline is to provide a common industry approach that may be used to effectively reduce the occurrences of leakage from potentially affected piping. This guideline is interim in that a final set of guidelines is planned upon project completion to supplement or replace this document.

The specific locations recommended for assessment and/or inspection in this guideline are those where cracking and leakage has been identified in nominally stagnant non-isolable piping attached to reactor coolant systems in domestic and similar foreign PWRs. Locations that are currently part of other augmented inspection programs as a result of specific cracking issues are excluded. Some of the piping that is covered by this guideline was previously identified as being susceptible to thermal fatigue with the issuance of NRC Bulletin 88-08 and its supplements. These locations are included since this guideline is based on the latest state of knowledge related to Bulletin 88-08 issues.

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 INTRODUCTION

This guideline presents evaluation and inspection recommendations for detecting potential cracking that might be occurring in normally stagnant piping systems attached to Pressurized Water Reactor (PWR) main reactor coolant piping. The guideline also provides criteria to identify lines that should not be susceptible to cracking. The objective of this guideline is to provide a common industry approach that may be used to effectively reduce the probability of leakage from piping potentially susceptible to thermal fatigue. This guideline is interim in that final guidelines are currently planned to be issued at project completion.

The specific locations recommended for evaluation and/or inspection in this guideline are those where cracking and leakage have been identified in domestic and similar foreign PWRs, and are not currently part of other augmented inspection programs as a result of specific cracking issues (e.g. B&W plant high pressure injection nozzle/thermal sleeves). Some of the piping that is covered by this guideline was previously identified as being susceptible to thermal fatigue with the issuance of NRC Bulletin 88-08 and its supplements [1].

Use of this interim guideline may result in recommended inspection of piping locations that are not included in ASME Section XI inservice inspection programs. However, the weld locations should already have been considered if the utility has implemented a risk-informed inservice inspection program. The piping systems selected for evaluation and potential inspection are those where actual pipe leakage events have occurred in operating PWR plants. Although there are relatively few locations expected to be susceptible to thermal fatigue that are not effectively managed by current programs, use of this guideline may assist plant operators in avoiding forced outages due to leakage during service. In addition, this guideline contains recommendations that may be useful in the implementation of risk-informed inservice inspection programs.

Section 2 of this guideline provides further background on formation of the PWR Materials Reliability Project Thermal Fatigue Issue Task Group (MRP TF-ITG) and this project. The project is summarized in Appendix A.

Section 3 of this guideline provides the specific recommendations for near-term assessment and possible volumetric examination. The piping systems and locations to be considered are those like drain lines and safety injection lines subject to valve inleakage. These are the ones where multiple cases of cracking and leakage have occurred in operating PWRs. In the case of safety injection lines subject to inleakage, effective programs may already be in place to assure that cracking will not occur as a result of responding to NRC Bulletin 88-08. This guideline provides no recommendation on frequency of inspection since an updated thermal fatigue management guideline is planned for issuance at project completion.

Section 4 summarizes the technical basis for the assessment/inspection recommendations. The locations to be addressed were taken from industry experience (Appendix B), supplemented by information developed in the EPRI Thermal Stratification, Cycling and Striping (TASCS) project completed in 1994 [2]. The reason for excluding evaluation of certain locations where leakage has been detected in some plants is also provided.

Section 5 describes how assessments may be performed to determine if volumetric examination and/or monitoring should be considered. Guidelines for conducting effective examinations and monitoring are also provided. Appendix C summarizes the evaluations performed at the EPRI NDE Center to develop effective examination techniques for detecting thermal fatigue cracking in small diameter stainless steel piping.

2 BACKGROUND

In 1987 and 1988, thermal fatigue cracking and leakage in several PWR plants resulted in the issuance of NRC Bulletin 88-08 [1]. In each of these events, the cracking was attributed to thermal cycling mechanisms not considered in initial plant design. The cracking was in nominally stagnant non-isolable lines attached to reactor coolant system piping. Later investigations showed that two of the three cases described in the Bulletin could be attributed to in-leakage of cold water toward the reactor coolant system. Interaction of this leakage with turbulence penetration effects from the reactor coolant piping resulted in cyclic conditions of hot and cold water on the inside of attached piping, eventually leading to thermal fatigue cracking and leakage. In the third case, the leakage was attributed to cyclic out-leakage from an isolation valve.

In 1995, leakage from a drain line in a PWR plant was attributed to the combined effects of turbulence penetration into the nominally stagnant un-insulated line and an inadequately designed support. More recently, there have been two additional similar pipe leakage events, one in a domestic plant and one in a foreign plant. These and other related events are summarized in Appendix B.

In 1997, cracking occurred in a B&W plant High Pressure Injection/Makeup line due to a loose thermal sleeve. Although this event was not in a nominally stagnant line, the potential effects of thermal fatigue cracking in small diameter safety injection lines became an issue.

In all of these cases, the occurrence of thermal fatigue cracking has not resulted in a pipe break, only leakage. However, the costs associated with evaluation, repair and plant unavailability have been significant. Based on a recent NRC-sponsored study related to fatigue effects during a 60-year license renewal operating period, thermal fatigue cracking does not have a significant contribution to core damage frequency [3]. Thus, the utility decision to assess the potential effects of thermal fatigue in non-isolable lines should be a balanced decision based on both plant safety and economics.

In early 1998, there were discussions between the nuclear industry and the Nuclear Regulatory Commission (NRC) regarding additional volumetric examination of Class 1 high pressure safety injection piping. Based on the instances of thermal fatigue cracking in these types of lines (at Oconee, Farley and several foreign plants), NRC concluded that surface examination was not sufficient to maintain the integrity of the reactor coolant pressure boundary piping throughout the design life. The industry position was that inspection of all safety injection piping was not required and that alternate evaluations and/or monitoring programs could provide adequate assurance that leakage would not occur. There was also a question as to the effectiveness of volumetric examination in detecting fatigue cracking in the smaller diameter piping systems.

As a result, an MRP ad hoc team was formed to investigate the issue and to provide recommendations to MRP utility management on how the industry could develop effective methods for managing thermal fatigue cracking in these lines. On March 10, 1999, creation of the Thermal Fatigue Issue Task Group (TF-ITG) was approved by the MRP Executive Group and Senior Representatives. In mid-1999, a preliminary program was described to the NRC consisting of tasks related to data collection, screening and analysis, inspection, monitoring, modifications and related tasks, culminating in final guidelines for thermal fatigue management in late 2001. As a result of NRC concerns that there were no more immediate licensee actions, it was committed that this set of interim guidelines would be made available by mid-2000.

The thermal fatigue project was approved by MRP executive management in August 1999. Contractors were chosen in late 1999. Many of the individual task reports are being prepared for the TF-ITG group review and approval concurrently with this guideline. A final thermal fatigue management guidelines document is planned to be issued upon project completion.

3 RECOMMENDATIONS

The following recommendations are provided for evaluation and possible inspection of lines where thermal fatigue has a demonstrated potential. The recommendations are based on the current state of knowledge and are not intended to prohibit the use of new knowledge or operating experience in making decisions about the need to perform further evaluation or inspection of certain lines.

The recommendations are provided for two specific types of locations as described in the following sections. Additional details on the basis for choosing these locations and the methods for assessment, inspection and/or monitoring are provided in Sections 4 and 5.

3.1 Lines with Potential In-Leakage

3.1.1 Identification of Lines

Valve in-leakage toward the reactor coolant system has the potential of causing cracking associated with the cyclic interaction between the leakage flow and turbulence penetration from the reactor coolant piping.

Included: The potentially affected lines are those such as high head safety injection lines which have a connection to the high pressure Chemical and Volume Control System (Charging) for Westinghouse- and CE-designed plants and to the Makeup System for B&W-designed plants. These are the only systems where there is a potential for continuous valve leakage toward the reactor coolant system. These were the lines identified in NRC Bulletin 88-08, where cracking occurred at the Farley and Tihange plants. On some plants (notably those of B&W and CE designs), direct in-leakage from the high pressure systems may not be possible.

Excluded: Lines that experience normal flow (charging or makeup) are not susceptible to valve leakage effects. Alternate charging lines and auxiliary spray lines, where the pressure driving force is much lower and there has been no identified cracking or leakage, are also excluded in this interim guideline.

3.1.2 Evaluation Criteria

Programs put in place since soon after NRC Bulletin 88-08 was issued may be used to show that there is no potential for valve in-leakage such that potentially affected lines can be excluded from further evaluation. Considerations can be given to the following or other factors:

- Presence of dual in-series isolation valves which would prevent leakage flow,
- Systems to limit or reduce the pressure in piping between the high pressure source and the check valve nearest the reactor coolant system piping,
- A monitoring or leakage trending program that periodically shows that isolation valves are not leaking or are leaking at a very low rate that has not lead to cracking in the past, and
- Periodic temperature monitoring of the piping.

Since there is a possibility of change in the leakage rate of isolation valves with time or operating cycles, any monitoring or leakage trending programs must assure that significant valve leakage does not develop during the life of the plant.

Figure 3-1 shows the logic used in evaluating whether a particular line is recommended for further evaluation or inspection. For most lines, it can be shown that no further evaluation or inspections are required.

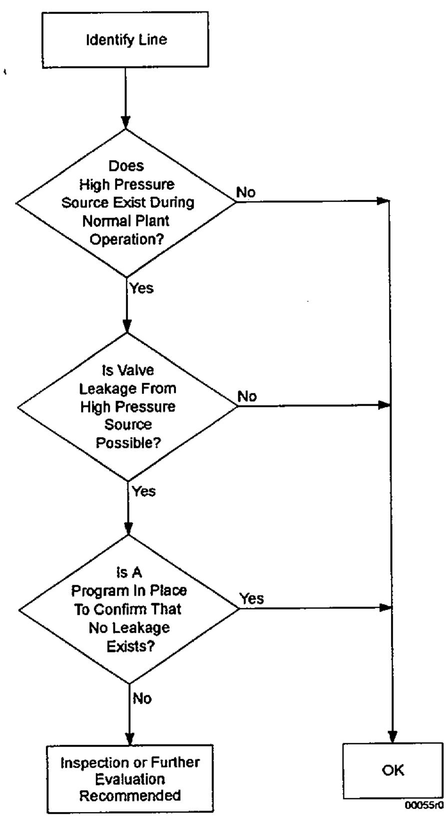


Figure 3-1
Logic for Evaluating Lines for Potential In-Leakage

3.1.3 Inspection Recommendations

For those lines that are potentially susceptible to in-leakage and might be affected by thermal fatigue, inspection is recommended. The potentially susceptible lines will be those attached to either the top or side of the reactor coolant system piping.

For butt-welded elbows (or bent pipe) to horizontal piping as shown in Figure 3-2, 100 percent circumferential volumetric examination (UT or RT) is recommended for the weld between the elbow and the horizontal pipe. The examination volume should also include the elbow (or bent pipe) base metal above the bottom of the adjacent horizontal pipe.

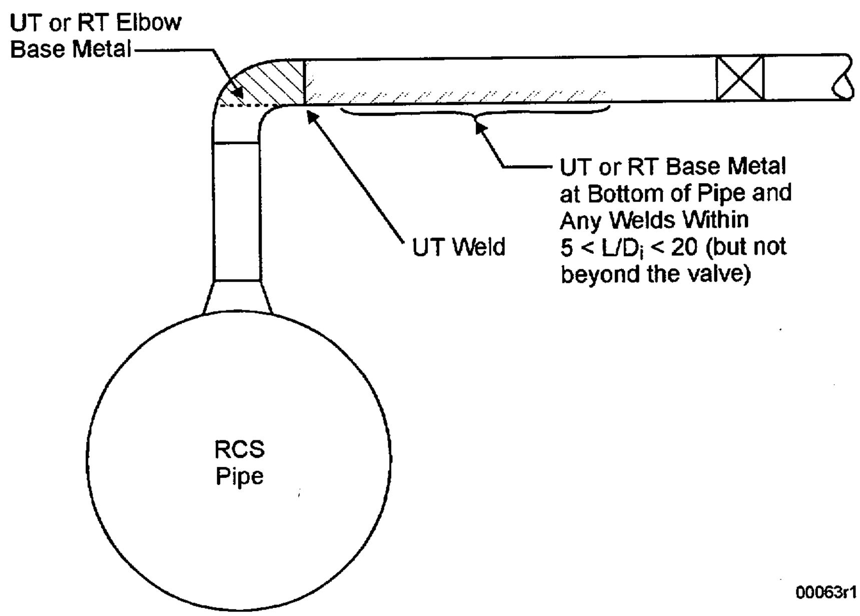


Figure 3-2
Examination Zones for Butt-Welded Lines Vertically Upward from RCS Piping

For socket-welded elbows to horizontal piping as shown in Figure 3-3, 100 percent circumferential volumetric examination (UT or RT) is recommended for the base metal within ½ inch of the toe of the socket weld on the horizontal piping.

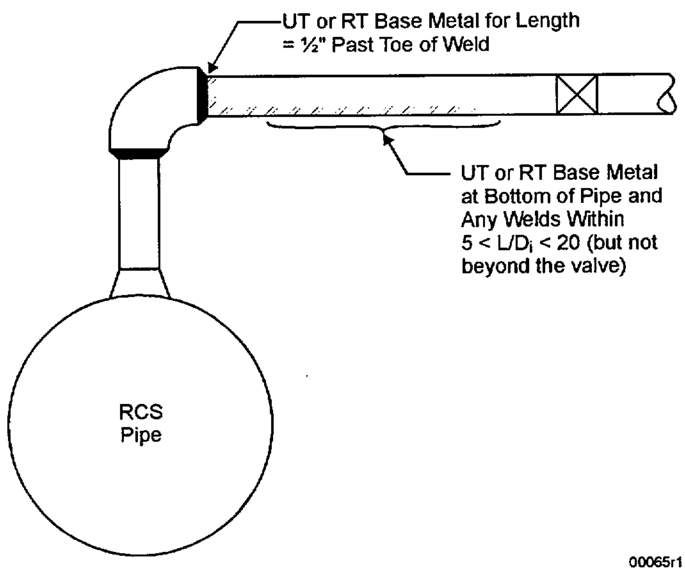


Figure 3-3
Examination Zones for Socket-Welded Lines Vertically Upward from RCS Piping

For piping attached to the side of the RCS piping, as shown in Figure 3-4, 100 percent volumetric examination of the pipe-to-nozzle (or safe-end) weld is recommended.

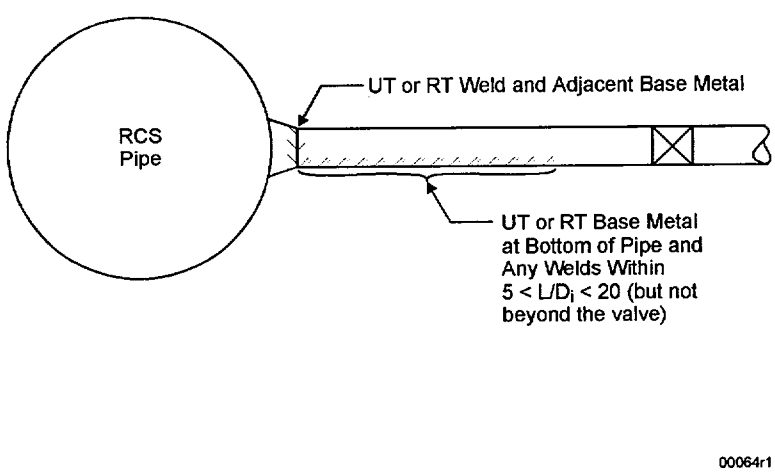


Figure 3-4
Examination Zones for Horizontal Lines from RCS Piping

For all horizontal piping as shown in Figure 3-2 to 3-4, including that beyond elbows, volumetric examination (UT or RT) is recommended for the base metal along a ½-inch wide zone at the bottom of the piping within the region defined by $5 < L/D_i < 20$ where L/D_i is defined in Section 5.1.5. If there are girth butt welds within the horizontal piping of this region, 100 percent circumferential volumetric examination (UT or RT) is recommended.

The complete length of piping ($5 < L/D_i < 20$) should be visually examined for evidence of leakage, except that examination of any piping or welds beyond the first isolation valve is not required. Guidelines on inspection volumes and effective inspection techniques are provided in Section 5.2.

3.2 Drain Lines/Excess Letdown Lines

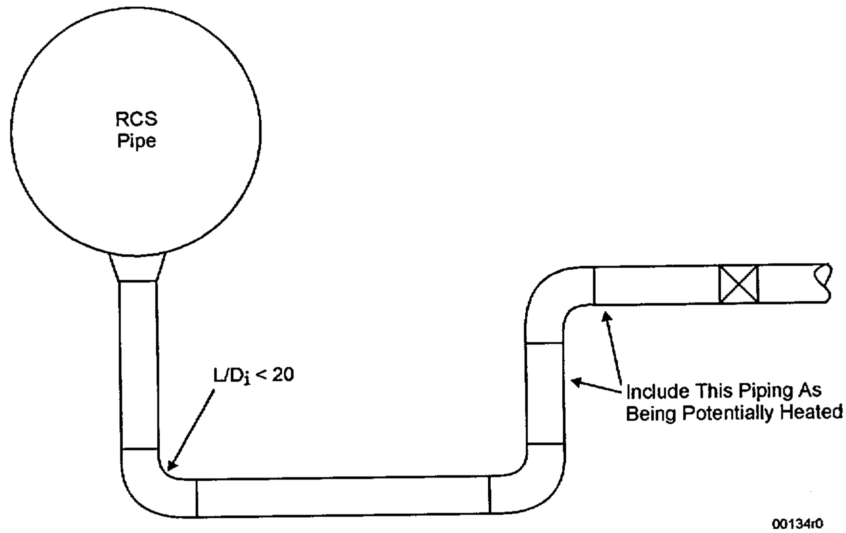
3.2.1 Identification of Lines

Potentially susceptible drain lines and excess letdown lines are normally stagnant lines that exit the bottom of reactor coolant piping and then turn horizontal prior to encountering an isolation valve. There have been recent cracking events in the first elbow or in the elbow to horizontal pipe weld at Oconee-1, TMI-1 and Mihama (in Japan). The cracking has been determined to be associated with cyclic turbulence penetration into the affected region interacting with relatively cooler water in the horizontal piping.

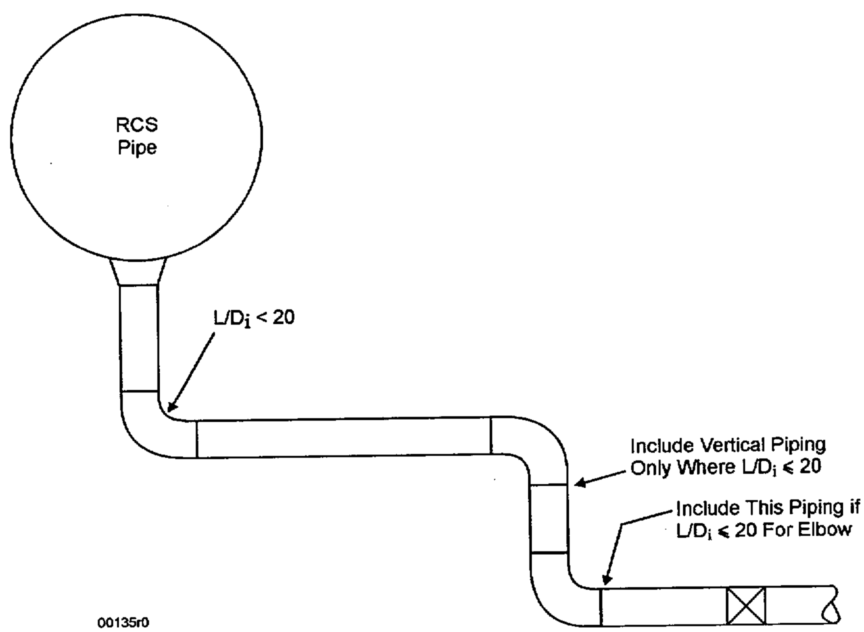
3.2.2 Evaluation Criteria

In order to experience cyclic thermal stresses, the first vertical-to-horizontal elbow must be within the turbulence penetration zone and there must be a source of water that is cooled by heat loss from the piping. Based on the plants where the failures occurred, either of the following criteria may be used to exclude a line from further evaluation or examination:

- Exclude lines where the first vertical-to-horizontal elbow is beyond $L/D_i = 20$, where L/D_i is defined in Section 5.1.5., or
- Exclude insulated piping where the length of potentially heated piping beyond the vertical-to-horizontal elbow is less than 10 feet to 1) an isolation valve, or 2) a downward run of piping beyond $L/D_i = 20$ that will limit the convection of heated water further into the piping system. This is illustrated in Figure 3-5 for piping that has both upward and downward facing piping beyond the first elbow. If the piping is not insulated, it could potentially run much colder due to much higher rates of heat loss. In this case, it should not be excluded.



a) Piping with Upward-Facing Segments



b) Piping with Downward-Facing Segments

Figure 3-5
Length of Drain/Excess Letdown Piping Potentially Heated by Effects of Turbulence Penetration

This logic is reflected in Figure 3-6.

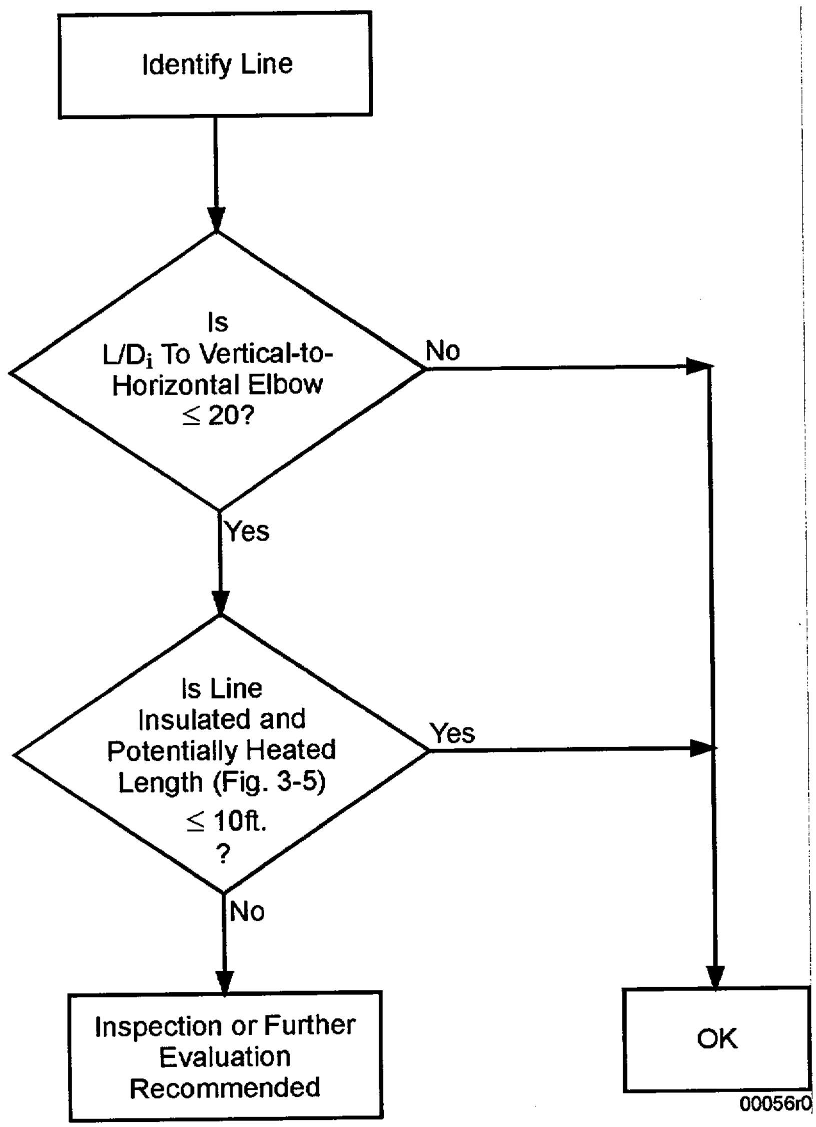


Figure 3-6
Logic for Evaluating Drain/Excess Letdown Lines

3.2.3 Inspection Recommendations

If the piping can not be excluded by the evaluation above, inspection or further evaluations are recommended. The following describes the regions that should be examined.

For butt-welded elbows (or bent pipe) to horizontal piping as shown in Figure 3-7, volumetric examination (UT or RT) is recommended 1) for the weld between butt-welded elbows and the horizontal piping, and 2) the base metal of the elbow that is below the top of the adjacent horizontal pipe.

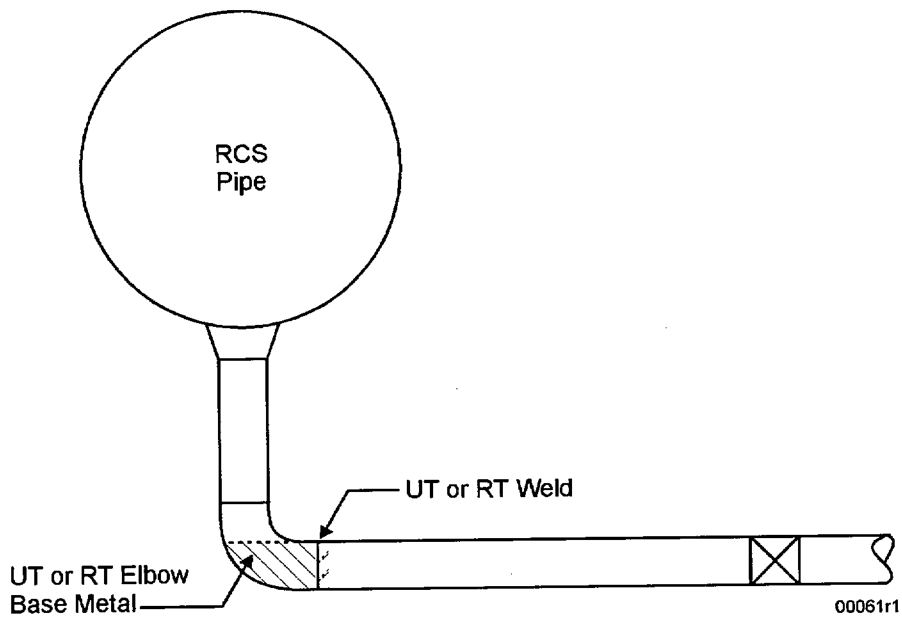


Figure 3-7
Examination Zones for Butt-Welded Drain Lines

For socket-welded elbows to horizontal piping as shown in Figure 3-8, volumetric examination is recommended for the base metal within ½ inch of the toe of the socket weld on the horizontal piping. In addition, the piping near the socket weld should be visually examined for evidence of any leakage.

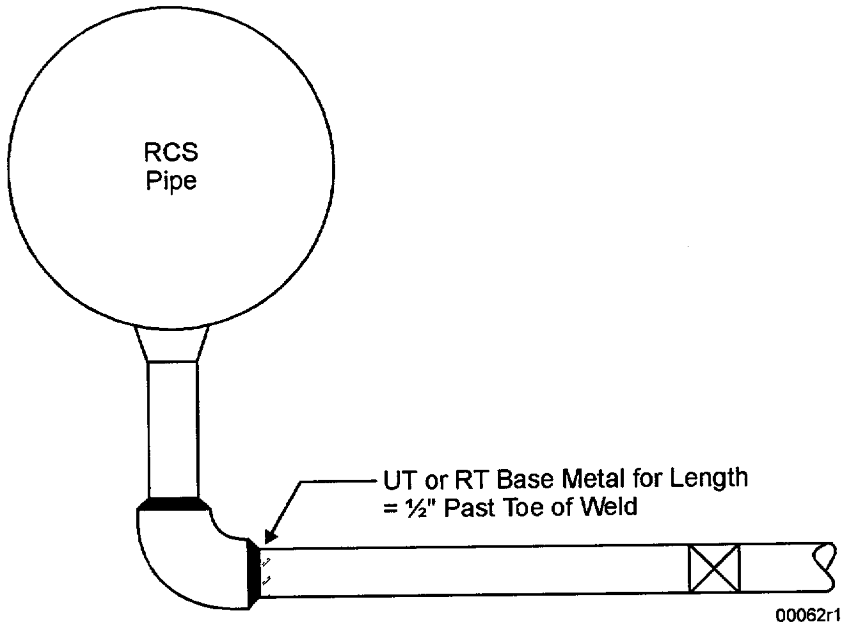


Figure 3-8
Examination Zones for Socket-Welded Drain Lines

4
DISCUSSION OF BASIS FOR CHOICE OF LOCATIONS

In developing this set of interim guidelines, locations were identified where there would be some relatively higher potential for cracking and leakage but where fatigue was not being managed through other industry programs. For other potentially affected locations, it was judged to be acceptable to recommend no further action until the final Thermal Fatigue Management Guidelines are published at project completion.

Thus, the following screening criterion was applied:

- Identify locations where thermal fatigue cracking/leakage of Class 1 piping inboard of the first isolation valve had occurred based on worldwide experience,
- Exclude those that appear to be unique in nature and not related to in-leakage or turbulence penetration effects, and
- Exclude those being managed through other industry programs

The remaining locations were those where additional plant-unique assessments and possible inspection are recommended.

4.1 Summary of Cracking Experience

Appendix B contains a list of cracking/leakage events for normally stagnant lines attached to PWR reactor coolant systems. There have been 14 reported events attributable to thermal fatigue cracking worldwide. An evaluation of each event is included for purposes of establishing interim thermal fatigue inspection/evaluation guidelines.

The review of the events shows that leakage events of the Farley type (with in-leakage from the high pressure charging pump source) and the TMI type (for downward facing drain lines) are the ones that have occurred with the most frequency. The other types of leakage are either unique or are managed by other industry programs.

4.2 In-Leakage Events

A common characteristic of several of the events was that there was a high pressure source of water (commonly the Chemical and Volume Control System) that leaked toward the reactor system, typically into safety injection systems. This potential leakage path does not exist in all plants. Thus, if there is the potential for this leakage, it is recommended that it be further evaluated. If detailed evaluation shows that leakage past a single isolation valve is likely, then inspection is recommended.

In these events, the leakage has been typically noted in the elbow above the reactor coolant piping where the piping turns to a horizontal orientation. The horizontal section allows the cold leakage to flow at the bottom of the pipe where it might interact with turbulence penetration from the hot leg.

From work performed in the EPRI TASCS program [2], it was observed that turbulence penetration might be active in the region $5 < L/D_i < 25$. (See Section 5.1.5 for definition of L/D_i.) For typical reactor coolant piping with 50 ft/sec velocity, the estimated turbulent penetration mixing zone length with stratified in-leakage layers was 12-23 diameters [2, Section 3.8]. Test data presented in the report [2, Section 5.4] showed that the turbulence penetration length might be as low as five inside diameters. Because the effect of and number of cycles of turbulence penetration would be expected to be less in the higher L/D_i range, the recommended range has been taken as $5 < L/D_i < 20$ for these interim guidelines.

The primary inspection zone has been taken as the elbow and the first horizontal pipe weld, based on the plant leakage experience. In addition, a bottom-of-the-pipe zone is included since this would be the most highly stressed zone if a thin layer of leakage flow cycled in the bottom of the pipe. Any welds in the horizontal piping are also included due to the potential for construction defects, residual stresses and stress risers that would be likely areas where cracking might initiate in the presence of thermal cycling.

4.3 Drain Line Events

For the cracking in the lines off the bottom of the RCS piping, no in-leakage was involved. However, the cyclic behavior of turbulence penetration energy transport into relatively cooler water in the horizontal piping below has resulted in through-wall leakage at several plants. The maximum turbulence penetration length has been taken as 20 inside diameters as discussed in 4.2. Thus, any lines off the bottom of the RCS piping with vertical length $L/D_i \geq 20$ may be excluded.

The inspection zone is chosen from the fatigue failure operating experience, and includes the elbow and the first horizontal elbow-to-pipe weld.

Note that there may be additional lines that are downward facing from the reactor coolant system piping that could potentially see the same effects but are excluded in these interim guidelines since no leakage has been observed:

- Small (≤ 1") sampling lines: These lines are generally of a size from which leakage is within the capability of the normal plant makeup system. These lines may be designed to the requirements of the ASME Code for Class 2 piping. Thus, these lines are excluded from evaluation and inspection in this interim guideline.
- Residual heat removal suction lines (also called shutdown cooling or decay heat removal suction lines): These lines are much larger in diameter. The one leakage event at Genkai was attributed to cyclic leakage from a valve packing leakoff line. Thus, these lines are excluded from further evaluation and inspection in this interim guideline.

5 GUIDELINES FOR ASSESSMENT, INSPECTION AND MONITORING

In this section, additional guidance is provided on how to evaluate whether inspections might be warranted and how to perform effective examination for thermal fatigue at locations recommended in Section 3.0. In addition, guidelines for monitoring in lieu of inspections are provided.

5.1 Assessment

The first step in determining if thermal fatigue is possible is to perform an evaluation of the reactor coolant system attached piping. The general approach was defined in Section 3.0. More detailed guidelines follow.

5.1.1 Evaluation of High Pressure Sources

In order to have in-leakage potential during normal operation, there must be a high pressure source of water and a path through which it can leak.

On Westinghouse (W) and Combustion Engineering (CE) designed plants, the charging pumps normally operate to inject water from the Chemical and Volume Control system back to the reactor coolant system through the charging lines. This is a normal path for flow back to the reactor coolant system. On some plants, the charging pumps perform a dual purpose and can also function as high pressure safety injection pumps. In this case, there is a flow path from the charging pumps to the safety injection nozzles.

On Babcock and Wilcox (B&W) designed plants, the high pressure source is from the makeup pumps, which also serve as high pressure injection pumps. Since the makeup pumps must run at a higher pressure than the reactor coolant system, a high pressure source exists.

For all PWR plants, there is a high pressure source. However, in order to have in-leakage, there must also be a leakage path from the high pressure source. If there is no possible leakage path (with a valve that might leak), then it may be concluded that there is no potential for leakage. In some plants, this path may not allow flow to occur during normal operation, due to multiple closed valves or pressure relief on intermediate headers.

5.1.2 Evaluation of Possibility of Valve Leakage

All valves have the potential for some leakage. In making the determination if valve leakage can cause in-leakage to the normally stagnant line adjacent to the reactor coolant system, there are several possible reasons that can lead to the conclusion that significant valve leakage cannot occur.

In some plants, there may be two normally closed isolation valves in the potential leakage path. The potential for significant leakage past two valves is considered to be low enough that this will be assumed to eliminate the possibility of leakage.

In order to have leakage into the reactor coolant system, there must be a pressure differential across the check valve in the piping adjacent to the RCS. If the pressure in the piping somewhere in the flow path between the high pressure source and the RCS nozzle is maintained at a pressure less than RCS pressure, then valve leakage cannot occur.

5.1.3 Monitoring

Another means of showing that no leakage is occurring is to perform monitoring. Where monitoring is used, the following monitoring frequency (during normal operating pressure conditions) is considered sufficient to show that leakage is not occurring:

- Following each startup from cold shutdown conditions, and/or
- Following each closure of the normal isolation valve.

Since leakage is not expected to change during plant service and the valves are not expected to be opened and closed during operation, this provides adequate assurance that leakage will not occur during operation. There are many means of monitoring to show that leakage does not occur. Effective monitoring programs are defined in 5.3.

5.1.4 Valve Leakage Trending

Many valves required to prevent in-leakage are included in plant leak rate testing programs. Although this testing may not be accurate enough to demonstrate zero leakage, trending of continued very low leakage rates is adequate proof that the potential for leakage is low enough to not cause thermal fatigue cracking. Where this approach is used, it is recommended that valve leakage trend testing be conducted following each refueling outage or other outage where the isolation valves have been opened.

5.1.5 Concept of L/D

In the determination of the potential for thermal cycling in branch lines and for determining the regions to inspect, the guidelines in 3.0 refer to L/D_i. In this term, L is the length from the inside face of the reactor coolant pipe to a location on the branch pipe and D_i is the branch line inside diameter. The piping length to the location includes both horizontal and vertical piping as

illustrated in Figure 5-1. It is relatively straightforward to calculate L/D_i for a straight piece of pipe. However, real piping systems may have elbows and can have changes in diameter.

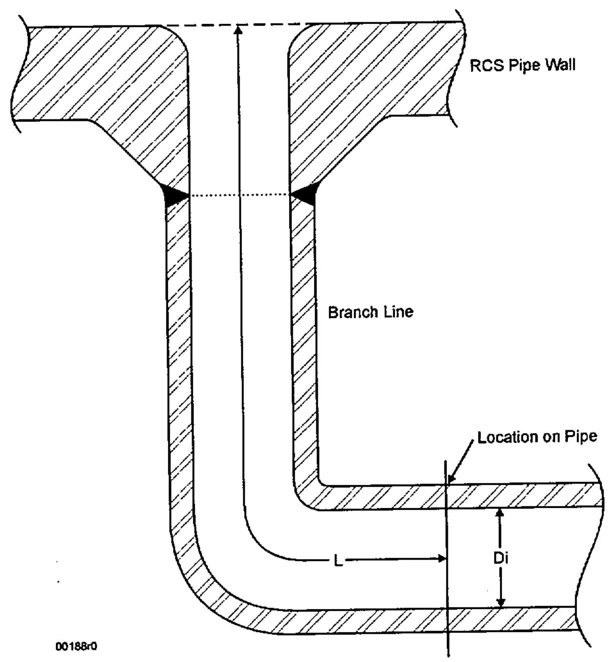


Figure 5-1
Illustration of L and D_i for a Branch Pipe

In the case of pipe runs with diameter changes, one must calculate an equivalent L/D_i. Consider the following case:

- 2-inch inside diameter pipe 14 inches long
- 2-inch to 4-inch reducer 4 inches long
- 4-inch inside diameter pipe 6 inches long
- $L/D_{eq} = \Sigma\,(L/D_i)$ (for all segments)
- $L/D_{eq} = 14/2 + (2/2 + 2/4) + 6/4 = 10$

For elbows, the length should be taken at the centerline length. For example, the length for a 4-inch long radius elbow (with nominal bend radius = 6 inches) would be $12\pi/4 = 9.43$ inches. In evaluating drain lines, one evaluation criteria is written in terms of the L/D_i to the vertical-to-horizontal elbow. In this case, L/D_i can be taken as the distance to the midpoint of the elbow. These definitions are somewhat arbitrary, but are provided to avoid ambiguity.

5.2 Inspection Guidelines

The MRP project for development of thermal fatigue nondestructive examination technology is briefly described in Appendix C with further details provided in Reference 4. The following provides further details for piping system inspection to meet the recommendations provided in Section 3.0, where evaluation shows that further inspection should be undertaken. Inspection of piping potentially affected by thermal fatigue cycling is best accomplished by ultrasonic examination (UT), although radiographic methods (RT) can also be used to detect significant cracking that exceeds the evaluation standards of ASME Section XI, IWB-3500. Radiographic methods did not typically detect cracking that was less than 10 percent of wall thickness [4].

5.2.1 General Examination Recommendations

Examination of small (<4-inch) diameter piping is more difficult than examination of larger diameter piping. Pipe wall curvature becomes a factor in backwall wave reflection and in assuring adequate contact between the transducer and the outer surface of the pipe. Examination of elbow base metal is even more difficult due to the complex curvature. On the other hand, it was demonstrated that it is possible to detect the occurrence of thermal fatigue cracking, although crack depth sizing is not sufficiently accurate to be reliable. Special UT transducers may be required on small diameter piping, especially elbows.

All potentially susceptible lines should be examined. However, it is not necessary that 100 percent of the recommended examination volumes be inspected. Experience shows that high cycle thermal fatigue cracking due to turbulence penetration and valve in-leakage effects is generally fairly wide spread. Thus, if full examination is not possible due to obstructions, weld crowns, etc., inspections that cover most of the recommended examination volume should adequately detect the presence of thermal fatigue cracking. To assure that examiners can detect thermal fatigue cracking, they must be properly trained. It is important for them to understand the types of cracking that they are trying to detect. High cycle thermal fatigue cracking is generally characterized by multiple initiation sites, the presence of crazing and fairly tight cracks, as compared to intergranular stress corrosion cracking (IGSCC), where significant branching occurs.

Thus, examiners should be trained. The recommended training is as follows:

- Previous formal qualification for piping ultrasonic examination such as the ASME Section XI Appendix VIII qualifications administered by the Performance Demonstration Initiative (PDI) or some other industry-recognized standard, and
- A special indoctrination (approximately 4 hours) to familiarize examiners with the peculiarities of examination for thermal fatigue damage (as compared to IGSCC examination) and for the geometric considerations specific to small diameter piping.

A computer-based training course is currently being developed by EPRI and will be available at about the same time this report is published. The information in Reference 4 can be used as the basis for utility-unique training courses.

5.2.2 Examination Procedure

Reference 4 contains a generic procedure for the ultrasonic examination of small-diameter piping. Visual examination can be performed using established utility procedures.

5.2.3 Inspection Locations

Section 3.0 contains sketches that define the locations and extent of the piping that should be examined. Figures 5-2 and 5-3 show recommended inspection volumes for welds, identical to that recommended for inspection of thermal fatigue in risk-informed inspection programs [5].

In safety injection lines, cracking has been observed in the elbow and adjacent weld between the elbow and the pipe. However, thermal cycling would most likely occur on the bottom of the piping if the check valve were far-removed from the RCS. Thus, the regions for inspection for butt-welded elbows (and bent pipe) in lines susceptible to in-leakage thermal fatigue cracking are as follows:

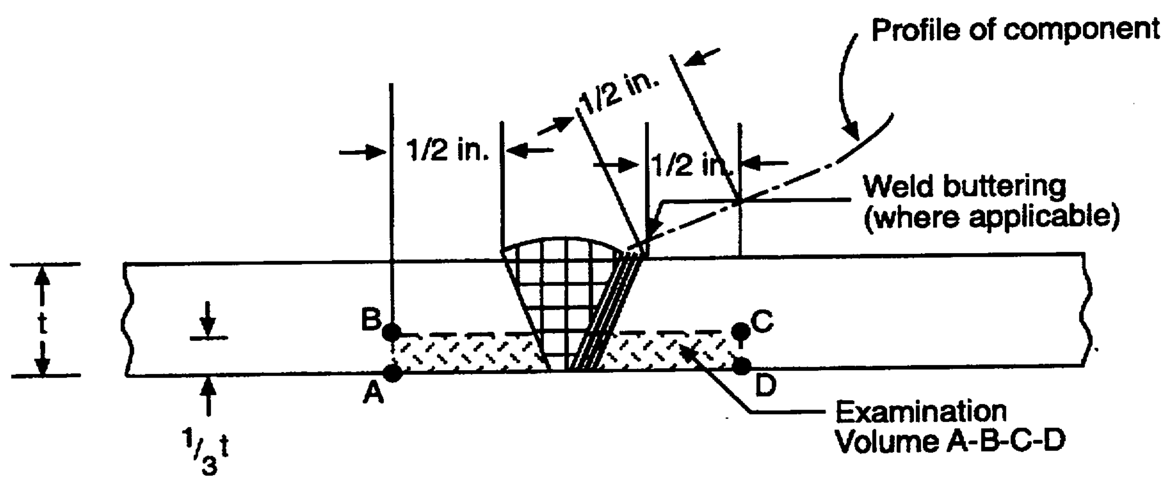


Figure 5-2
Examination Volume for Thermal Fatigue Cracking in Piping Welds Less than NPS 4

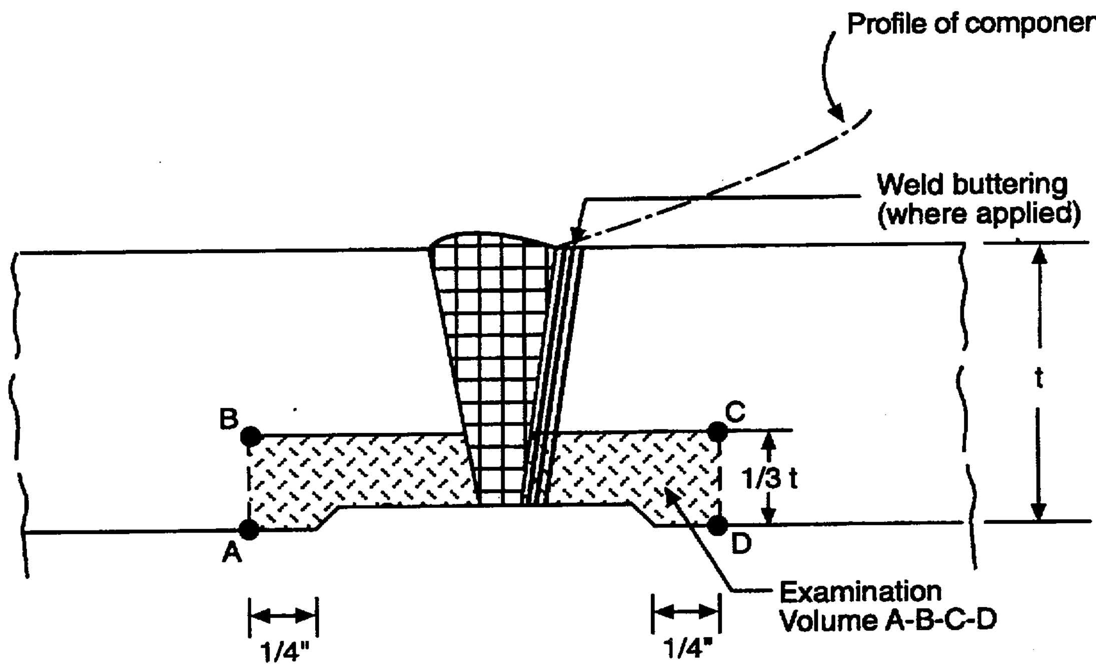


Figure 5-3
Examination Volume for Thermal Fatigue Cracking in Piping Welds NPS 4 or Larger

- The elbow base metal that is above the bottom outside diameter of the adjacent horizontal pipe segment should be examined. This examination should be conducted in such a manner that would pick up the existence of crazing or deeper circumferential, axial or skewed cracking on the inner surface.
- The weld between the elbow and the horizontal pipe segment of butt-welded elbows should be examined. This should include the additional amount of metal adjacent to the weld on the pipe side of the weld as shown in Figures 5-2 and 5-3. The expected cracking would be circumferential.
- An examination should be conducted along a line on the bottom of the pipe to detect the presence of craze cracking. The axial extent should be out to $L/D_i = 20$ but not beyond the check valve in the line. For ultrasonic examination, this may consist of axial line scans with the ultrasonic beam directed in both the upstream and downstream directions that cover a width of piping equal to about ½ inch. Radiography may be performed such that this region is fully examined.
- Any additional circumferential welds in the horizontal piping should also be examined as defined in Section 3.0 to detect the presence of circumferential cracking in the region of the weld. This would include the pipe-to-valve weld if it were within the $L/D_i \leq 20$ zone.
- There is no need to examine any piping beyond the check valve.
- In any regions where examinations indicate the potential existence of cracking or crazing, sufficient supplemental examinations should be performed to demonstrate whether flaws exist or not.

- No examinations are recommended in any vertical piping or in welds between vertical pipe and elbows.

For lines susceptible to in-leakage cracking with socket weld elbows, the recommendations are similar to the above. However, since ultrasonic examination has not yet been qualified for socket weld examination, only the region within ½-inch of the toe of the socket weld needs be examined. Due to the physical size of typical ultrasonic search units, the circumferential scans for axial and skewed cracking will only be capable of examination of that portion of the pipe base metal beyond about ¼ inch from the toe of the weld. Likewise, axial scans for circumferential cracking can only be applied from the pipe side, and not the weld side. The ultrasonic scan for the remainder of the piping is the same as for butt-welded piping. Radiographic examination can be performed for the entire area. The elbow and the socket welds should be visually examined for any evidence of prior leakage.

For lines which branch horizontally from the RCS piping and are susceptible to in-leakage cracking, the inspection recommendations are the same as for the horizontal piping butt-welded elbows, and include a scan of the bottom of the pipe and all welds (with adjacent base metal).

For drain lines (and other similar lines with diameter greater than one inch) that branch off the bottom of the RCS piping, the first elbow should be examined if the line is potentially susceptible to thermal fatigue. The examination recommendations are similar to those described above for bent pipe, butt-welded or socket-welded elbows, except that the elbow base metal below the top of the adjacent horizontal pipe outside surface defines the required examination volume.

In case there are geometries that are not specifically described above, the same philosophy for examination locations and volumes can be used to develop line-specific inspection guidelines.

5.3 Monitoring

Various types of monitoring may be used to assure the absence of in-leakage or thermal cycling.

5.3.1 General Monitoring Criteria

Monitoring can consist of several different approaches. For systems subjected to in-leakage, the monitoring can show that leakage does not occur (measuring flow, pressure, valve leakage, etc.), or can be based on detection of the thermal effects of leakage at the affected locations (temperature sensors). For drain lines, the monitoring must be targeted toward temperature sensing on the affected locations.

In-leakage occurs because an isolation valve is not closing tightly enough to prevent leakage past the seat. Since the amount of leakage could change with time (mainly as affected by open/close cycles), any monitoring to detect leakage or the effects of leakage must not be discontinued after monitoring results indicate that leakage is not occurring. It is recommended that data be taken and evaluated following each heatup from cold shutdown or after each open/close cycle of the isolation valve.

For drain lines, or other similar lines on the bottom of the RCS piping, monitoring need not be continuous. It is sufficient to take data during normal plant operation. If there is no evidence of any thermal cycling, then the instrumentation may be removed.

5.3.2 Monitoring as an Inspection Alternative

Following the assessments described in 5.1, locations may be identified as candidates for inspection. Installation of monitoring as an alternative to inspections can be considered. However, discovery of significant thermal cycling with the monitoring could indicate the potential for thermal fatigue or cracking. At the time of this interim guideline, methods have not yet been finalized in the MRP thermal fatigue project to predict the fatigue effects of thermal cycling and any evaluations of data remain the responsibility of the plant owner.

Thus, inspection is recommended for locations if they are potentially susceptible to thermal cycling, as determined by the evaluation techniques above, and there has been no previous monitoring to assure that thermal cycling is not occurring. It is not recommended that monitoring be initiated without a baseline inspection.

5.3.3 Temperature Monitoring

Temperature monitoring can be accomplished be using strap-on thermocouples or resistance temperature detectors. For monitored locations, it is recommended that the sensors be located at the top and bottom of the piping, sufficiently insulated to avoid effects from the surrounding environment. The following guidance is provided for placement of sensors:

- For horizontally oriented piping with elbows (or bent pipe) going downward to the RCS piping and subject to in-leakage, it is recommended that the sensors be placed on the horizontal elbow-to-pipe weld (or adjacent to the weld for socket-welded piping). Additional sensors may be placed at the check valve-to-pipe weld. The elbow sensors would be expected to detect cycling at the location where through wall cracks have been observed. Sensors at the check valve will detect the presence of leakage.
- For piping that exits the RCS piping horizontally, it is recommended that the sensors be placed at the check valve to detect leakage effects. Additional sensors could be placed on the line to detect cycling (e.g., at $L/D_i \sim 10\text{-}12$).
- For drain lines, it is recommended that a single pair of sensors be placed at the top and bottom of the elbow-to-horizontal pipe weld (or adjacent to the weld for socket-welded piping).

Additional sensors may be installed for purposes of redundancy. Two or three additional sensors could be equally spaced between the top and bottom sensors at each location on one side of the pipe to provide additional data for analytical evaluation.

Temperature monitoring can reveal either thermal cycling or steady stratification or both:

- Thermal cycling indicates the presence of turbulence penetration either 1) interacting with in-leakage in lines off the top or side of RCS piping, or 2) carrying hot water into colder

regions below the RCS piping in the case of drain lines. Thermal cycling would be expected to repeat with a frequency on the order of once per hour or less and could lead to high cycle fatigue.

- Steady stratification is typically seen in upward facing or horizontally attached lines that do not experience inleakage and is the result of natural convection cells forming due to heat loss from the enclosed fluid. For piping geometries that are beyond the turbulence penetration zone, large top-to-bottom temperature differences may develop. The contribution to thermal fatigue cracking is much less, however, because 1) the temperature gradients are generally less severe, and 2) the number of cycles is generally comparable to those for heatup and cooldown.

To evaluate temperature monitoring data, any top to bottom temperature cycling that is less than 50°F is considered to be acceptable, regardless of the rate of cycling. Methods and criteria for evaluation of monitoring results are currently under development and are planned to be included in the final thermal fatigue management guidelines. In the interim, evaluation of monitoring results must be developed by the plant owner for each situation.

For un-insulated drain lines, thermography during normal operation may be utilized to show that conditions are steady and cold. Any evidence that turbulence penetration is heating the un-insulated piping to more than 50°F above local ambient temperature should be further evaluated and may require additional monitoring or inspection.

5.3.4 Pressure Monitoring

Pressure monitoring may be used for lines potentially susceptible to in-leakage. The region between the isolation valve and the check valve should be monitored with a pressure instrument with sufficient accuracy to show that the piping outboard of the check valve is at least 5 psi less than the minimum RCS loop normal operating pressure (including consideration of measurement uncertainties). This may require periodic pressure reduction in the region, since this region probably normally runs at reactor coolant pressure (since the check valves tend to leak much more than the isolation valves).

5.3.5 Leakage Monitoring

Leakage monitoring is a means of demonstrating that there is insufficient leakage to cause in-leakage thermal cycling. This can be accomplished by installing a bleed off line in the region downstream of the isolation valve and physically measuring the amount of leakage that occurs with full ΔP across the isolation valve. Determination of an acceptable leakage rate must be based on plant-unique evaluations.

6
FEEDBACK REQUEST

6.1 Implementation of these Guidelines

If, during the course of implementing the recommendations in this guideline, it is determined that the guidelines cannot be achieved or that meaningful results are not obtained, it is requested that feedback be provided to the MRP TF-ITG through EPRI. Sufficient details should be provided such that alternate approaches can be developed. Improved guidelines based on these notifications, as well as results of the ongoing TF-ITG project, will be contained in the final Thermal Fatigue Management Guidelines to be issued upon project completion.

6.2 Assessment of Fatigue Susceptibility

It is requested that summaries of results of analytical assessments or results of monitoring based on these guidelines be provided to the TF-ITG (through EPRI) since they may be of assistance in development of the final thermal fatigue management guidelines.

6.3 Inspection Results

It is requested that results of inspections be provided to the TF-ITG (through EPRI). It is anticipated that a formal procedure for providing these to EPRI and for posting on the EPRI Thermal Fatigue Web Site will be included with the final thermal fatigue management guidelines.

7 REFERENCES

1. United States Nuclear Regulatory Commission, Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems" 6/22/1988, including Supplements 1, 2, and 3, dated 6/24/88, 8/4/88 and 4/11/89.

2. TR-103581, *Thermal Stratification, Cycling and Striping (TASCS),* EPRI, Palo Alto, CA, March 1994.

3. NUREG/CR-6674, "Fatigue Life Analysis of Components for 60-year Plant Life," Pacific Northwest National Laboratory for U.S. Nuclear Regulatory Commission, June 2000.

4. TR-100152, *NDE Technology for Detection of Thermal Fatigue Damage in Piping* (MRP-TF-06), EPRI, Palo Alto, CA, September 2000.

5. TR-112657, Revised *Risk-Informed Inservice Inspection Evaluation Procedure*, Rev. B-A, EPRI, Palo Alto, CA, December 1999.

A
THERMAL FATIGUE PROJECT OVERVIEW

The MRP Thermal Fatigue project is composed of 13 tasks. The following summarizes each of the tasks, describing the outcome of each. Figure A-1 shows the inter-relationship between the tasks and how the outcome of each task can be used by utilities in assessing the potential for thermal fatigue or performing additional inspections, monitoring, maintenance, or plant modifications.

Task 1 – Project Management

Description: This is the overall effort by EPRI to manage the project.

Outcome: This assures that the project can be completed on time and within the budget established by the utilities.

Task 2 – Presentation of Final Report to NRC

Description: This represents the completion of the project. The NRC is being kept aware of the program progress through continued involvement in the MRP TF-ITG meetings.

Outcome: This task will attempt to assure that the final product of the project can be utilized by utilities in making decisions concerning issues of thermal fatigue.

Task 3 – Industry Operating Experience

Description: Review of existing industry databases and literature to capture details of leaks or identified flaws. Industry experience and the sharing of information of events that do not meet some higher reporting threshold could provide added value and new insights.

Outcome: A database that collects utility thermal monitoring results and observations which do not reach the level of reporting through LERs or other means. The database will contain historical experience and will make it available for future utility use. The vision is being able to collect information on lower level thermal anomalies that would indicate beyond-design thermal fatigue mechanisms that have the potential to cause thermal fatigue damage.

Task 4 – Thermal Fatigue Screening

Description: A simple screening model will be developed. The screening model will describe the thermal fatigue phenomena, factors necessary for thermal fatigue to occur, and the logic process for making this determination. The methodology will have a technically defensible basis and will be validated against known instances of thermal fatigue failures.

Outcome: Methodology for determining when and where significant thermal fatigue damage may occur in PWR piping systems.

Task 5 – Thermal Fatigue Monitoring Guidelines

Description: This task will provide guidance in the following areas:

- basis for implementing a monitoring program
- identification of state-of-the-art monitoring technology
- effective placement of monitoring sensors
- interpretation of monitoring data
- basis for discontinuing monitoring

Outcome: Practical guidance on the use of monitoring to detect potential thermal fatigue phenomena.

Task 6 – NDE Inspection Guidelines

Description: This task will assemble previous guidance on NDE methodologies (such as RT or UT) and make recommendations for specific NDE technology and variables. Recommendations will be made on the appropriate qualification of NDE examiners and procedures. The recommended means of evaluating NDE data and reporting levels will be provided. Contacts will be made internationally to determine any difficulties reached in applying NDE for thermal fatigue and laboratory investigations to verify performance. Research will be conducted to determine capabilities for producing thermal fatigue cracks and then mockups will be designed and fabricated. A set of mockups with individual cracks due to thermal fatigue will be needed, along with mockups containing thermal craze cracking.

Outcome: Guidance on NDE methodologies and recommendations of specific NDE technology and variables to use when inspecting for suspected thermal fatigue damage.

Task 7 – Plant O&M Guidelines

Description: This task focuses on how O&M practices can lead to thermal fatigue damage and on identifying recommended changes to eliminate the damage potential. PWR operating experience will be used to identify plant practices and corrective actions implemented at affected plants.

Guidance will be developed for use by utility engineers to aid in identifying O&M practices which may lead to fatigue damage and what actions may be taken to mitigate the consequences.

Outcome: Guidance for use by plant personnel to identify how O&M practices can create or minimize the potential for causing thermal fatigue events.

Task 8 – Thermal Fatigue Evaluation

Description: This task will develop more rigorous analysis guidelines to aid the engineer in evaluating thermal fatigue situations.

Outcome:

- A simplified thermal fatigue evaluation methodology will be developed to assist the engineer in estimating fatigue usage.
- An analysis guide, documenting the techniques and describing comprehensive methodologies for analytical reconciliation of thermal fatigue phenomena using or based on ASME Section III methodology, will be developed for more rigorous treatment of thermal fatigue.

Task 9 – Plant Modification Guidelines

Description: This task will identify and describe plant modifications that can be successful in avoiding the potential for thermal fatigue.

Outcome: Identification of cost effective plant changes that would eliminate need for future monitoring and piping augmented inspections.

Task 10 – International Technical Exchange

Description: This task will support identification of and possible participation in important foreign R&D activities that could contribute to resolution of the thermal fatigue issue. An international workshop on thermal fatigue experience and R&D was conducted by EPRI in mid-2000.

Outcome: Awareness of and access to foreign information of value to US utilities for detection, assessment, mitigation, and prevention of thermal fatigue damage.

Task 11 – Thermal Fatigue Management Guidelines

Description: This task represents the principal product of this ITG. It assembles the results of the other tasks, documents conclusions drawn from that work, and provides recommendations for managing thermal fatigue.

Outcome: The "TFMG" will be a compilation of methods for assessment, screening, monitoring, analysis, and remediation for and management of thermal fatigue. The ITG will seek NRC staff acceptance of the guideline.

Task 12 – Develop and Deliver Training

Description: This task develops and delivers training for utility engineers and others in applying the "TFMG". This task is aimed at utility personnel (operations, maintenance, and engineering) to increase their overall knowledge and awareness of cyclic thermal fatigue and how plant operations and maintenance may be contributors to the phenomena.

Outcome: More knowledgeable and experienced personnel and more effective management of thermal fatigue issues.

Task 13 – Monitor ASME Section XI WG Changes

Description: This task monitors the activities of the ASME Section XI Working Group on Inspection Systems and Components. Inspection guidance developed by this ITG will be reviewed with appropriate Code groups to determine if such guidance should form the basis for other Code revisions.

Outcome: Coordination of ITG and ASME Code activities

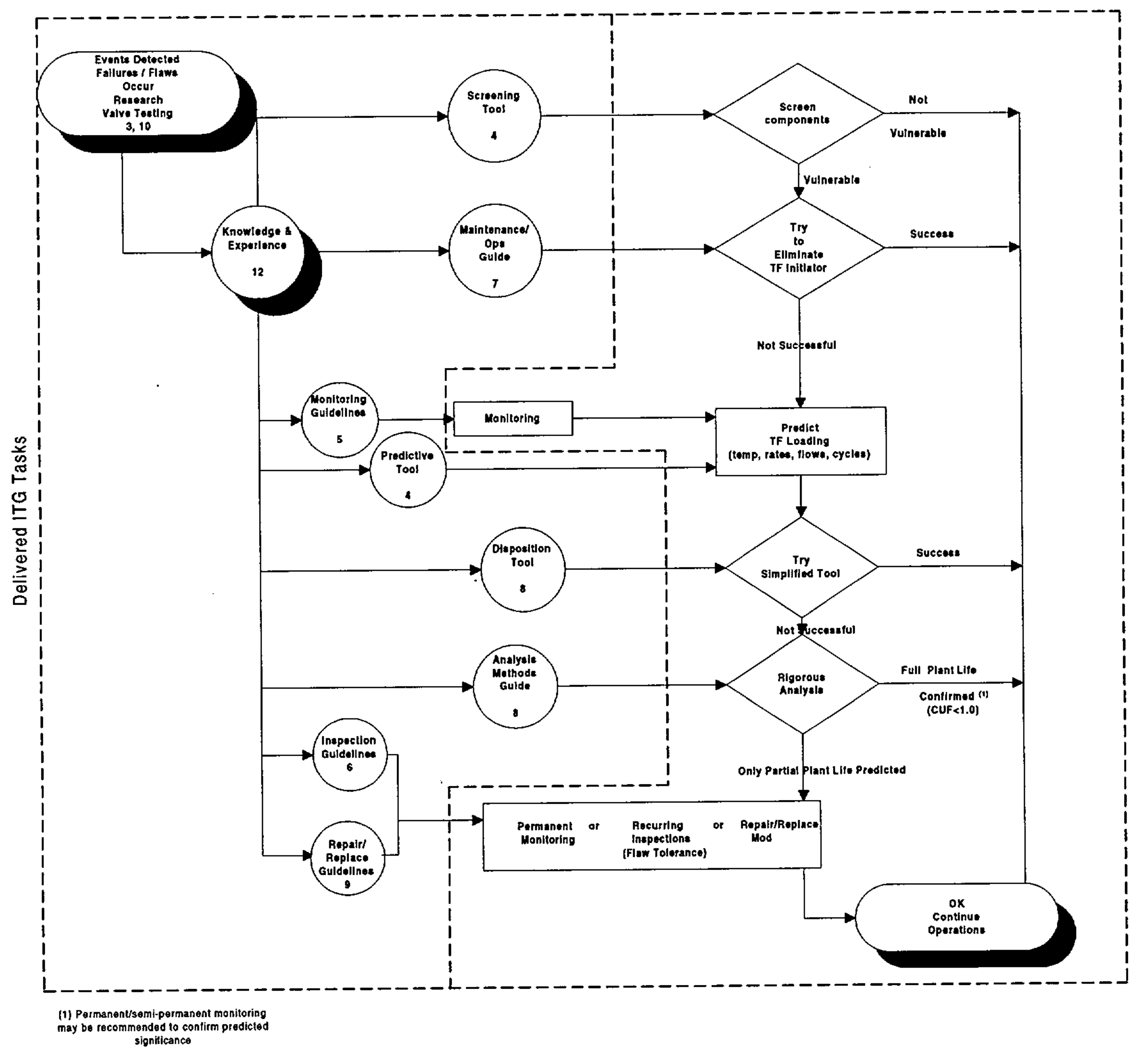


Figure A-1
Project Flowchart

B
DESCRIPTION OF SIGNIFICANT THERMAL FATIGUE CRACKING/LEAKAGE EVENTS

Table B-1 shows the relevant events where leakage was observed in reactor coolant systems in PWR plants worldwide. The list includes only those events where the leakage is in non-isolable sections and the leakage is attributed to thermal fatigue effects. There are 14 events total. Additional details may be found in the Task 3 report [B.1].

For purposes of establishing interim thermal fatigue inspection/evaluation guidelines, the following evaluates each one and assesses its relative importance from the standpoint of near-term cracking potential.

Crystal River 3

This leakage was in the HPI/Makeup line and was caused by failure of a thermal sleeve. The design is unique to the B&W plant design and is being managed by an alternate program for B&W plants that assesses the integrity of the thermal sleeves. Because this program is in place, no further interim actions are recommended.

Obrigheim

This leakage was attributed to cold flow toward the reactor coolant system from a high pressure source (Chemical and Volume Control System). It is related to a cold injection line that is a feature not found in domestic PWR plant designs.

Farley Unit 2

This leakage was attributed to cold flow toward the reactor coolant system from a high pressure source (Chemical and Volume Control System) through safety injection piping. This event resulted in the issuance of NRC Bulletin 88-08.

Tihange 1

This leakage was very similar to that which occurred at Farley 2.

Genkai 1

The leakage in this foreign PWR was a one-of-a-kind situation and was attributed to cyclic flow from an RHR suction isolation valve leakoff line. It was addressed in NRC Bulletin 88-08. Due to the uniqueness of this event, no interim action is recommended.

Dampierre 2

This event is very similar to the Farley 2 leakage event.

Loviisa 2 (Spray System)

This event was caused by stratification in a valve body with a unique design and was attributed to interaction between the auxiliary spray and main spray. Since there have been no instances of similar failures in plants in the U.S., no interim action is recommended.

Biblis B

This event appears to be similar to the Obrigheim event and was associated with in-leakage of cold injection water from the Chemical and Volume Control system. The leak is attributed to a design feature that is not found in U.S. plants.

Three Mile Island 1

This leakage was from a stagnant drain line below the RCS cold leg. The elbow was approximately 12 inside diameters in length below the reactor coolant system, where the drain line turned horizontal. Except near the nozzle, the line was not insulated. Cracking was attributed to cyclic turbulence penetration into the relatively colder line in the elbow and horizontal piping.

Dampierre 1

This leakage event was very similar to the occurrence at Farley 2.

Loviisa 2 (Drain)

This event was attributed to cross-leakage in a line connecting the RCS cold leg and hot leg drains. Cyclic flow was attributed to cyclic thermal expansion of the valve internals. Because it is not clear that the same type of lines exist in plants in the US and there has been no similar event here, no interim action is recommended.

Oconee 2

This event was very similar to that occurring at Crystal River. Thermal fatigue management is part of a B&W plant program to monitor the thermal sleeve integrity. Therefore, no further interim action is recommended.

Mihama 2

This leakage occurred in a nominally stagnant excess letdown line and is quite similar to the TMI 1 drain line event. In this case, the line was insulated but was approximately 18 feet from the elbow to the first isolation valve.

Oconee 1

This leakage was very similar to that which occurred at TMI 1.

Summary

Review of these events indicates that leakage events of the Farley type (with in-leakage from the high pressure CVCS source) and the TMI type (drain lines) are the most frequent. The other types of leakage are either very unique or are managed by other industry programs.

Reference

1. 1001006, *Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant Systems* (MRP-25), EPRI, Palo Alto, CA, November 2000.

Table B-1
PWR Reactor Coolant Leakage in Non-Isolable Lines Attributed to Thermal Fatigue

Plant	Event Date	Initial Criticality Date	NSSS Vendor	Piping System	Description of Cracking	
					Location	Size
Crystal River 3[1]	1/82	1/77	B&W	Makeup/High Pressure Injection	Weld between check valve and safe end	140-degree circumferential crack; two crack initiation sites: one on the inside surface and one on the outside surface
Obrigheim[2]	6/86	9/68	Siemens	Chemical and Volume Control	Weld between RCS nozzle and first elbow	Crack extended 2.75 inches circumferentially at the inside surface, 0.5 inches at the outside surface
Farley 2[3]	12/87	5/81	W	Safety Injection	Heat affected zone of elbow-to-pipe weld	Crack extended 120 degrees circumferentially at the inside surface, 1 inch long at the outside surface
Tihange 1[3]	6/88	2/75	ACLF	Safety Injection	Elbow base metal	3.5 inches long at the inside surface, 1.5 inches long at the outside surface
Genkai 1[4]	6/88	1/75	MHI	Residual Heat Removal	Heat-affected zone of elbow-to-pipe weld	Crack extended 3.8 inches circumferentially at the inside surface, 0.06 inches at the outside surface
Dampierre 2[3]	9/92	12/80	Framatome	Safety Injection	Check valve-to-pipe weld and base metal of straight pipe	Crack extended 4.3 inches circumferentially at the inside surface, 1.0 inches at the outside surface
Loviisa 2[5]	5/94	10/80	AEE	Auxiliary Spray Line	Pressurizer auxiliary spray line control valve body	Crack extended 3.1 inches along the horizontal surface and 1.0 inches along the vertical surface of the valve body
Biblis-B[2]	2/95	3/76	Siemens	Chemical and Volume Control System	Base metal of straight pipe and weld between pipe and a tee	Crack extended 2.0 inches axially at the inside surface, 0.8 inch at the outside surface

Table B-1
PWR Reactor Coolant Leakage in Non-Isolable Lines Attributed to Thermal Fatigue (Continued)

Plant	Event Date	Initial Criticality Date	NSSS Vendor	Piping System	Description of Cracking	
					Location	Size
Three Mile[6] Island 1	9/95	6/74	B&W	Cold Leg Drain Line	Weld between a 90-degree elbow and horizontal line	Crack extended 2 inches circumferentially at the inside surface, 0.55 inches at the outside surface
Dampierre 1[3]	12/96	3/80	Framatome	Safety injection	Base metal of a straight portion of the pipe	The crack extended 3.1 inches circumferentially at the inside surface 0.9 inches at the outside surface
Loviisa 2[7]	1/97	10/80	AEE	Hot Leg Drain Line	Weld between a T-joint piece and a reducer	65-degree circumferential crack, 1 inch long
Oconee 2[1]	4/97	11/73	B&W	Makeup/High Pressure Injection	Safe-end to pipe weld	Crack extended 360° circumferentially at the inside surface, about 77° circumferentially on the outside surface
Mihama 2[6]	4/99	4/72	MHI	Excess-letdown line of chemical and volume control	Base metal of first elbow below cross-over leg	1 inch long on the inside surface, 0.25 inches long on the outside surface
Oconee 1[6]	2/00	4/73	B&W	Cold Leg Drain	Elbow base metal	0.5" long on the inside surface and 3/16" long on the outside surface

Notes on cause of thermal fatigue cracking:

1. B&W plant loose thermal sleeve in MU/HPI nozzle (2)
2. Hot/Cold water mixing unique to Siemens design (2)
3. Valve in-leakage/turbulence penetration (4)
4. Cyclic valve out-leakage (1)
5. Thermal cycling internal to pressurizer spray valve (1)
6. Drain or excess letdown line turbulence penetration (3)
7. Loop-to-loop cross flow due to leaking valve (1)

C SUMMARY OF EPRI NONDESTRUCTIVE INSPECTION DEVELOPMENT PROGRAM

Background

Piping failures of thermal fatigue origin prompted a review of examination practices currently used, particularly in small-diameter piping (less than 4-inches in diameter). Because volumetric examination techniques are difficult and most examinations of small-diameter piping are currently limited to surface examinations, a program was developed and completed that includes criteria and methods to improve the reliability of small-diameter piping examinations.

Program Results

The research conducted at the EPRI NDE Center resulted in significant improvements in the state of knowledge for thermal fatigue crack detection in small diameter piping. It is described in the MRP TF-ITG report for Task 6.0. [C.1] This report reviews the relevant international experiences, describes evaluation results of candidate nondestructive evaluation (NDE) technologies for thermal fatigue crack detection and, based on the results obtained, recommends guidance for NDE examination. The NDE techniques evaluated included ultrasonics, time-of-flight diffraction, radiography, ultrasonic spectroscopy, pulsed eddy current, and vibro-modulation.

From the results obtained, the following trends and observations were derived:

- Manual ultrasonics detected and length sized the craze and thermal fatigue cracks in all the mockup samples tested.
- The time-of-flight diffraction technique detected, length sized and depth sized the craze and thermal fatigue cracks in five of the six mockup samples tested. The craze cracking was not detected in one sample.
- Conventional radiography detected and length sized all the thermal fatigue cracks and some of the craze cracks in the mockup specimens. Detection was limited to flaws deeper than 10 percent of the wall thickness.
- Ultrasonic spectroscopy detected the craze cracks in four of the six mockups evaluated. The area extension of the craze cracking was accurately sized. The thermal fatigue cracks were not detected.
- Pulsed eddy current detected the craze cracks with depths greater than 10 percent of the wall thickness. The technique did not detect the thermal fatigue cracks.

- Vibro-modulation detected the presence of craze cracks in two of the three mockup specimens tested. Areal extension of the craze cracking could not be derived from the signals.
- Conventional eddy current was judged to be not suitable for this application because of the limited penetration of the current field.

From these results the following conclusions were obtained:

- Manual ultrasonics was found to perform best among the technologies considered in this evaluation. This technique is viable for detecting and length-sizing cracks of thermal fatigue origin in small bore piping when applied in accordance with the EPRI procedure.
- Time-of-flight diffraction is a viable detection technique for scanning large areas semi-automatically to be supplemented with ultrasonics when more precise length-sizing evaluations can lead to a better repair schedule.
- Conventional radiography with fine-grain film is a viable technique for detecting thermal fatigue cracks deeper than 10 percent of the wall thickness. This technique can be supplemented with manual ultrasonics when the examination objective includes detecting craze damage in its early stages.
- Ultrasonic spectroscopy, pulsed eddy current, and vibro-modulation require further development before they can be implemented in a power plant environment for thermal fatigue crack inspection.

In accordance with the above, a generic procedure for inspection using manual ultrasonics was developed. The generic procedure was further tested on a field-extracted 1 ½ -inch diameter elbow that exhibited thermal fatigue damage.

Finally, it was recommended that examiners receive approximately 4 hours of indoctrination prior to performing examination in power plants. This indoctrination should be administered to examiners who have previously demonstrated proficiency in ultrasonic examination of piping welds through some industry standard. A computer-based training module is being developed to allow utilities to effectively implement this indoctrination. This module is expected to be available in early 2001.

For full details of the program and the generic procedure mentioned above, obtain the referenced final report. [C.1]

Reference

1. 1000152, *NDE Technology for Detection of Thermal Fatigue Damage in Piping* (MRP-23), EPRI, Palo Alto, CA, October 2000.

Target:
Nuclear Power

About EPRI

EPRI creates science and technology solutions for the global energy and energy services industry. U.S. electric utilities established the Electric Power Research Institute in 1973 as a nonprofit research consortium for the benefit of utility members, their customers, and society. Now known simply as EPRI, the company provides a wide range of innovative products and services to more than 1000 energy-related organizations in 40 countries. EPRI's multidisciplinary team of scientists and engineers draws on a worldwide network of technical and business expertise to help solve today's toughest energy and environmental problems.

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