Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-XXNP)

E217858NP (Draft)

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PRODUCT DESCRIPTION

The Materials Reliability Program (MRP), formed in January 1999, is an association of utility and industry representatives focusing on pressurized water reactor (PWR) vessel, material and related issues. The Fatigue Issue Task Group was formed to evaluate the potential effects of thermal fatigue on normally stagnant piping systems attached to reactor coolant system (RCS) piping as they might be affected by valve in-leakage and/or turbulence/swirl penetration effects. This report provides needed guidelines and other good practice recommendations for evaluating and inspecting regions where there may be potential for thermal fatigue cracking that could lead to leakage and forced plant outages. It has been prepared to meet the objectives of Nuclear Energy Institute (NEI) 03-08, the industry materials initiative promulgated through the auspices of NEI.

Background

In 1998, the Nuclear Regulatory Commission expressed concerns that the surface examinations of small diameter (≤4-inch NPS) high pressure safety injection piping required by ASME Code, Section XI were not adequate and that volumetric examination should be considered. This led to formation of the MRP Fatigue Issue Task Group (ITG) to provide evaluation and assessment techniques that would identify if additional inspection or monitoring would lead to significant increases in piping system reliability.

Objective

The objective of this guideline is to provide a common industry approach to be used for assessing the potential for thermal fatigue cracking in normally stagnant non-isolable RCS branch piping. It expands upon the recommendations of an interim guideline (MRP-24) issued in 2000 and provides recommendations for an ongoing fatigue management program in affected lines.

Approach

The MRP thermal fatigue project is a multi-tasked effort to provide screening, evaluation, monitoring, inspection, operations, maintenance and modification guidance to enable utilities in avoiding thermal fatigue cracking due to valve in-leakage or turbulence/swirl penetration effects in affected lines. A set of interim guidelines was developed in year 2000 to provide timely feedback to utilities to foster an awareness of key locations where thermal fatigue cracking was most likely to occur. Non-destructive examination methods were developed that could be used to detect thermal fatigue cracking and crazing. Workshops were held at essentially all domestic plants to provide awareness and assess key piping systems for the potential for thermal fatigue effects. Continued efforts in the area of thermal-hydraulic testing and model development have been undertaken to improve methods for analytical evaluation and mitigation approaches for the subject lines.

Results

From the continued efforts, improved assessment criteria are provided that will allow utilities to determine if normally stagnant non-isolable branch lines attached to reactor coolant system piping might be affected by thermal fatigue resulting from valve in-leakage and/or turbulence/swirl penetration effects. Based on this assessment, lines may be shown as unaffected or specific locations may be identified for inspection or further evaluation. If susceptible locations are identified, the guidance provided herein shall be used to perform effective nondestructive examinations, to implement monitoring, or to take other actions to effectively manage thermal fatigue. A thermal cyclic load definition methodology is provided as a basis for continued operation based on fatigue usage and/or crack growth analysis.

EPRI Perspective

This guideline provides utilities with a set of evaluation criteria to assist in identifying locations potentially susceptible to thermal fatigue conditions not known at the time of initial plant design. Use of these guidelines, along with the associated nondestructive examination and evaluation techniques, will contribute to effectively managing thermal fatigue and assist in avoiding reactor coolant leakage and unplanned outages due to thermal fatigue cracking.

Keywords

Fatigue
Thermal Fatigue
Thermal Cycling
Thermal Stratification
Inspection
Non-destructive Examination
Reactor Coolant Piping
NRC Bulletin 88-08
Cracking
Leakage

EXECUTIVE SUMMARY

In 1988, as identified in NRC Bulletin 88-08, there were several instances of thermal fatigue cracking in normally stagnant lines attached to reactor coolant system (RCS) piping. These cracks eventually resulted in leakage. This issue was addressed by utilities by conducting evaluations and monitoring to assure that further leakage would not occur. However, there were additional instances of thermal fatigue cracking in lines not specifically addressed in Bulletin 88-08. 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 or equal to 4-inch nominal pipe size. Based on several instances of thermal fatigue cracking in safety injection lines (at Farley, Oconee and several foreign plants), the 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 thermal fatigue cracking would not occur. In addition, the effectiveness of volumetric examination to detect fatigue cracking for these smaller diameter piping systems had not been established.

In March 1999, the Materials Reliability Program (MRP) Executive Group and Senior Representatives approved formation of the Fatigue Issue Task Group (ITG). In mid-1999, a program was described to the NRC consisting of tasks related to screening and analysis, monitoring, modifications and related activities, to culminate in final guidelines for thermal fatigue management. In 2000, an interim guideline was provided for evaluating safety injection lines and drain lines, systems that had experienced multiple industry occurrences of thermal fatigue leakage. The conclusions from the document were utilized in conducting workshops at essentially all domestic operating pressurized water reactor (PWR) plants.

Continued efforts in the area of thermal-hydraulic testing and model development were undertaken to provide improved methods for assessing the potential for swirl penetration thermal fatigue, to develop analytical evaluation methods for affected lines, and to provide a comprehensive management approach. This revised guideline presents enhanced assessment methods to be used in determining if thermal fatigue cracking in normally stagnant piping systems attached to PWR reactor coolant system (RCS) piping can occur. Where the potential for fatigue degradation exists, recommendations for monitoring or further analytical evaluation are provided that shall be used to determine if further actions (e.g., maintenance, repair, monitoring and/or non-destructive examination) are necessary to assure that thermal fatigue cycling is not significant. The objective of this guideline is to provide a common industry approach to use for effectively reducing the occurrences of cracking in potentially affected piping.

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1 INTRODUCTION AND BACKGROUND

1.1 Introduction

This guideline presents screening, evaluation and inspection recommendations for assessing potential thermal fatigue cracking due to swirl penetration and/or valve in-leakage that may occur in normally stagnant non-isolable piping systems attached to pressurized water reactor (PWR) main reactor coolant system (RCS) 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 to use in effectively reducing the probability of cracking in and leakage from piping potentially susceptible to thermal fatigue. 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].

In an interim version of this guideline [2], some specific piping locations were recommended for evaluation and/or inspection where cracking and leakage had been identified in domestic and similar foreign PWRs, including the safety injection lines identified in Bulletin 88-08.

Use of this guideline may result in required inspection of piping locations that are not included in ASME Code, Section XI inservice inspection programs. However, the weld locations should already have been considered if a risk-informed inservice inspection program has been implemented. Use of this guideline will 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 1.2 of this guideline provides further background on the PWR Materials Reliability Program (MRP) Fatigue Issue Task Group (ITG) formation and the thermal fatigue project history. The deliverables of the individual project tasks are summarized in Appendix A. Section 1.3 discusses individual plant owner compliance requirements in meeting this guideline.

Section 2.0 provides methodology for evaluating the potential for thermal fatigue in the RCS-attached lines. Criteria are provided that show some lines to be unaffected by the thermal fatigue mechanisms addressed herein. For the lines that might be affected, an evaluation methodology is provided to perform a fatigue assessment. For these lines, further monitoring, non-destructive examination, loads mitigation or physical modification/replacement may be required to assure that future unacceptable fatigue cracking will not occur.

Section 2.0 is sub-divided into six parts. Section 2.1 discusses the overall assessment and management methodology and refers to the other sub-sections for details. Section 2.2 discusses the details of screening and evaluation; Section 2.3 discusses effective monitoring; Section 2.4

discusses effective inspections; Section 2.5 discusses mitigation options; and Section 2.6 provides guidance for conducting a stratification analysis.

Section 3.0 summarizes the requirements of this guideline and compares them to the interim guideline [2]. Although this guideline imposes requirements for evaluation of additional lines, an assessment of a large sample of plant configurations and programs in place was performed during workshops at each utility site. Essentially all plants have addressed the cold-water in-leakage issue for safety injection lines following issuance of NRC Bulleting 88-08. Programs exist at most plants for managing this issue. However, an assessment of the effectiveness of those programs shall be performed to ensure that the requirements of this final guideline are met. In addition, re-evaluation of bottom connected normally stagnant RCS-attached lines (e.g., drain lines and residual heat removal suction lines) may identify the need for additional fatigue management actions.

1.2 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 normally stagnant non-isolable lines attached to RCS piping. Investigations by EPRI in the thermal stratification, cycling, and striping (TASCS) project to investigate these events 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 [3]. Interaction of this leakage with turbulence/swirl penetration effects from the reactor coolant piping resulted in cyclic conditions of hot and cold water on the inside of the attached piping, eventually leading to thermal fatigue cracking and leakage. In the third case, the leakage was attributed to cyclic out-leakage past a normally closed valve.

In 1995, leakage from a drain line in a PWR plant was attributed to the combined effects of turbulence/swirl penetration into the normally 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.

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 normally stagnant line, the potential effects of thermal fatigue cracking in small diameter safety injection lines became an issue.

These and other related events are summarized in Appendix B. A complete evaluation of thermal fatigue leakage events was completed in this project and documented in Reference 3. In none of these cases has the occurrence of thermal fatigue cracking resulted in a pipe rupture, however, leakage has occurred. For each of the events, the costs associated with evaluation, repair and plant unavailability have been significant. Although the results of a recent NRC-sponsored study related to fatigue effects during a 60-year license renewal operating period indicate that thermal fatigue does not have a significant contribution to core damage frequency [4], the utility decision to assess the potential effects of thermal fatigue in non-isolable lines should be a balanced decision based on both economic and plant safety considerations.

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), the 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.

In March 1999, the MRP Executive Group and Senior Representatives approved formation of the Thermal Fatigue Issue Task Group (later renamed the Fatigue Issue Task Group or Fatigue ITG). In mid-1999, a preliminary program was described to the NRC. The program consisted of tasks related to data collection, screening and analysis, inspection, monitoring, maintenance, modifications to plant systems or plant operations, and preparation of a final guideline for thermal fatigue management. Because of NRC concerns that there were no immediate licensee actions, an interim guideline document (MRP-24) was made available in January 2001 [2].

Following issuance of the interim guideline, additional testing and evaluations were undertaken to better understand the thermal fatigue mechanisms that had been responsible for cracking in the non-isolable normally-stagnant branch lines [5, 6]. This testing has allowed models to be developed that can be used to predict line susceptibility with the methodology developed and benchmarked against known leakage events [7]. This guideline now presents enhanced assessment methods to be used in determining if thermal fatigue cracking in normally stagnant piping systems attached to PWR main reactor coolant piping can occur. Where the potential for this component degradation exists, recommendations for monitoring or further analytical evaluation are provided that shall be used to determine if further actions (e.g., maintenance, repair, monitoring and/or non-destructive examination) are necessary to assure that thermal fatigue cycling is not significant.

1.3 Compliance Responsibilities

This document has been prepared under the MRP program as defined in MRP-130 [8]. It is structured per the objectives and requirements of the Nuclear Energy Institute (NEI) materials initiative [9]. As such, all owners of PWR reactors shall perform an evaluation of their plants utilizing this guideline within two years after the publication date. Needed actions as identified in Section 3.0 shall be addressed by this evaluation. If these initial evaluations lead to additional actions to assure that thermal fatigue cracking does not occur (e.g., more detailed evaluations, inspections, additional monitoring, etc. as outlined in this guideline), these actions shall be taken in a timely manner, consistent with outage schedules.

2 RECOMMENDATIONS AND LINE ASSESSMENT METHODOLOGY

Branch lines are generally attached to the top or bottom of RCS loop piping, but some lines intersect at the side or other orientations. The cyclic thermal stratification occurring within such piping is affected by line orientation such that the thermal cycling mechanisms may be broadly classified based on line geometry. This classification results in two generic configurations: the top or side connected lines (up-horizontal/horizontal, (UH/H) configuration); and the bottom connected lines (down-horizontal (DH) configuration). The classification of attached piping is based on geometric considerations, with the mechanisms for thermal cycling fundamentally different in each. Common to each is the existence of a swirling vortical flow structure in the branch line induced by the high velocity flow in the RCS piping which decays with distance from the RCS. Thermal stratification and cycling in UH/H configurations is caused by the interaction of this swirl flow with in-leakage of cold water from a leaking normally closed valve. In contrast, no valve in-leakage is necessary in DH configurations, where thermal cycling occurs due to the cyclic penetration and retreat of the swirl flow in the branch line, combined with heat transfer to the environment. See Section 2.2 for additional discussion and Reference [7] for a more detailed discussion of the phenomena. Figures 2-3, 2-4, and 2-5 illustrate the UH, H, and DH configurations.

Section 2.1 discusses the overall assessment and management methodology for this thermal stratification and cycling and refers to the other sub-sections for details regarding screening and evaluation, monitoring, inspections, and mitigation options.

2.1 Assessment of Lines

Plant operators shall perform an assessment of all normally stagnant lines attached to the reactor coolant piping that are greater than one-inch nominal diameter to determine if actions are required to prevent thermal fatigue cracking. These lines shall include at least the following lines on PWR plants:

- Safety injection lines
- Charging lines that are not in service (either permanently or on a cycle-by-cycle basis) – generally Westinghouse plants
- Drain lines
- Excess letdown lines (if this line exists in a specific plant configuration)
- Residual heat removal (RHR) suction lines, (also called decay heat removal (DHR) lines for B&W plants and shutdown cooling (SDC) suction lines for CE plants)
- Any other normally stagnant lines with nominal diameter greater than one inch.

The assessment approach is shown in Figure 2-1 for lines that are connected to the upper circumferential arc or side of reactor coolant piping. Figure 2-2 shows the assessment approach for lines connected to the lower circumferential arc of RCS piping, such as drain lines, excess letdown lines, or RHR suction lines. Sections 2.1.1 through 2.1.4 provide the details corresponding to the elements in the two figures.

As shown in Figure 2-1 and Figure 2-2, the key activities relate to the following:

- Screening: Use of simple rules or evaluations to show that lines are not susceptible.
- Evaluation: Use of models described or referenced in this report to determine 1) the significance of thermal cycling, 2) the cumulative fatigue usage factor, or 3) predicted crack growth.
- Inspection: Based on the results of screening and evaluation, determination of appropriate inspection locations and intervals.
- Repair/Replacement/Mitigation: If the potential exists for thermal fatigue cracking that cannot be resolved by evaluation/monitoring and inspection, other actions may need to be taken.

If there are plant modifications that affect these lines, the guidance provided herein shall be considered in implementing the plant change.

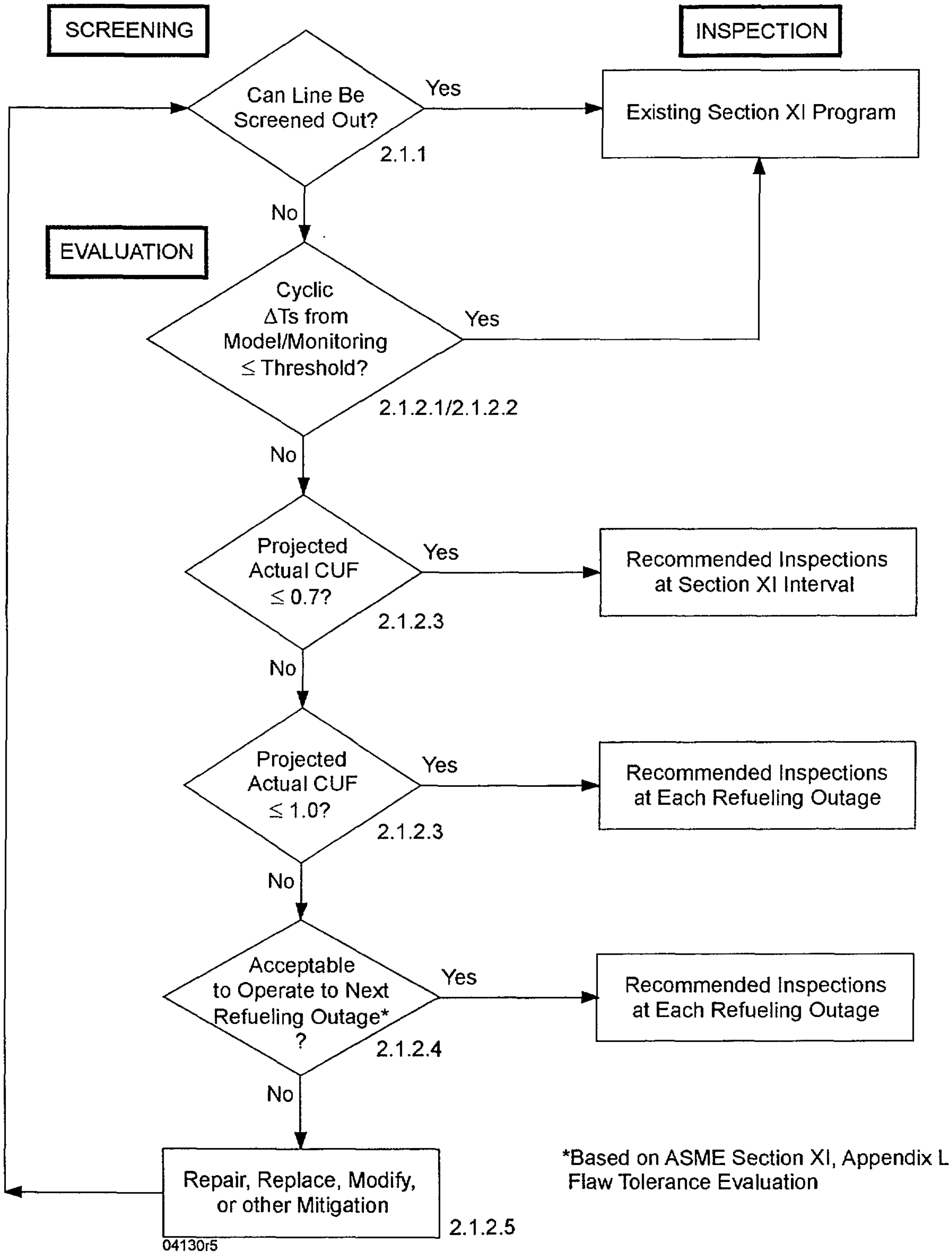

Figure 2-1
Assessment Approach for Side and Top Connected Lines

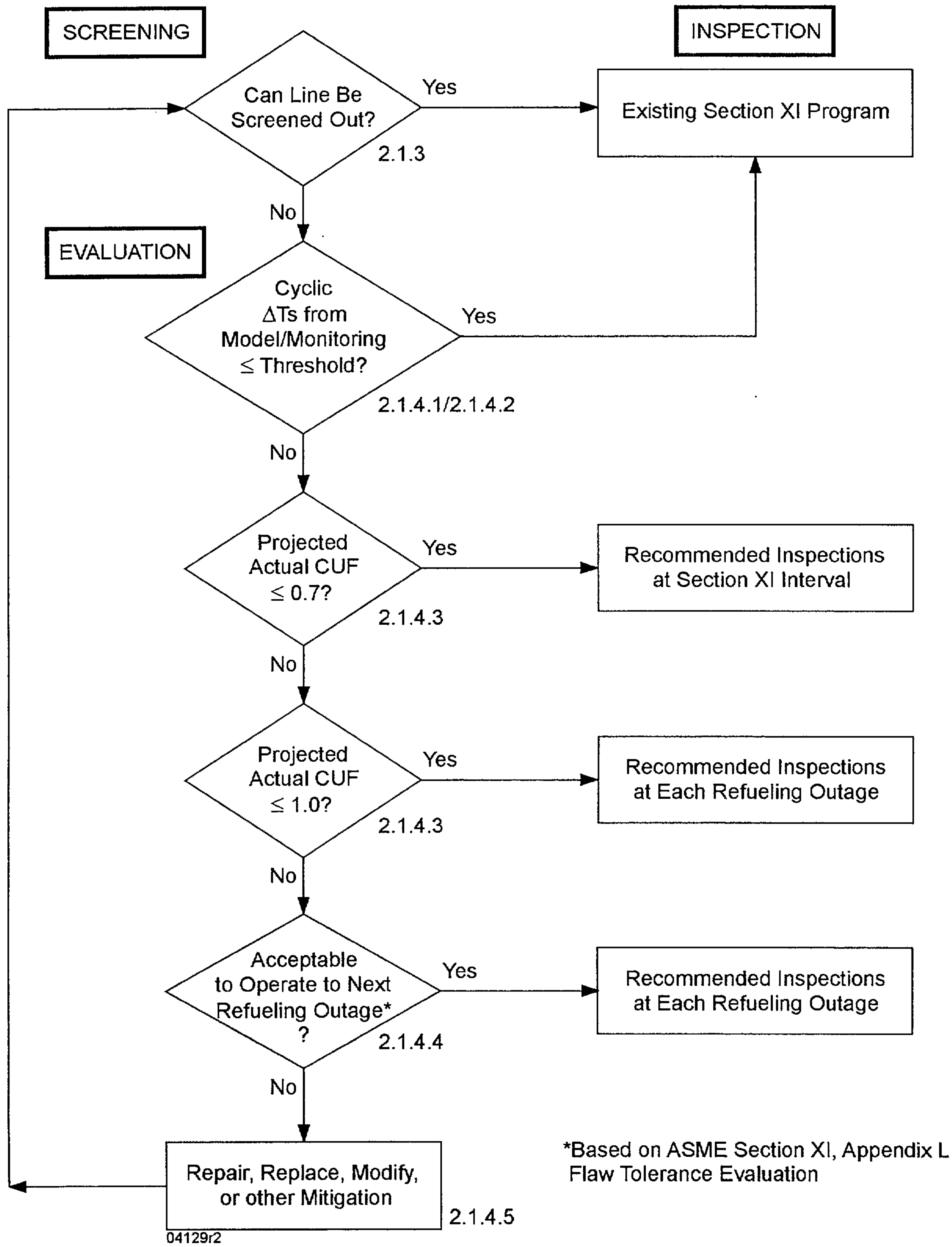

Figure 2-2
Assessment Approach for Bottom Connected Lines

2.1.1 Screening of Top or Side Connected Lines (UH or H)

The first step is to identify all normally stagnant branch lines attached to the RCS piping with a potential for in-leakage from a high-pressure source toward the RCS piping. These lines will typically include the Westinghouse plant Safety Injection (SI) and out-of-service (alternate) charging lines, and the B&W plant Emergency High Pressure Injection (HPI) lines. For most Westinghouse plants, both normal and alternate charging lines are provided, with the flow routed through one or the other during alternate operating cycles. Any line that is normally stagnant during some plant operating cycles must be included as a normally stagnant line in the assessment.

These lines will all have a horizontal section that interfaces directly into the RCS piping from the side or a horizontal section that turns downward then intersects the RCS from the top; hence they are referred to as horizontal (H) or up-horizontal (UH) lines in this report.

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If the line can be screened out using these criteria, no special examination of the lines is required and no further evaluations are needed. Inspection requirements remain unchanged from those of the current plant inservice inspection (ISI) program.

2.1.2 Evaluation of Top or Side Connected Lines (UH or H)

For lines that cannot be eliminated from consideration based on the screening rules of Section 2.1.1, two avenues are provided for initial evaluation.

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2.1.3 Screening of Bottom Connected Lines (DH)

The first step is to identify all normally stagnant branch lines attached to the bottom half of the RCS piping. These lines will typically include all drain lines, excess letdown lines, and the lines used for decay heat removal (residual heat removal suction lines for Westinghouse plants, shutdown cooling suction lines for Combustion Engineering plants, and decay heat removal suction lines for B&W plants).

These lines will all intersect the RCS piping at the bottom (or within the lower circumferential arc) of the RCS piping before turning horizontal; hence they are referred to as down-horizontal (DH) lines in this report.

Thermal fatigue will not occur and no further evaluation is required if either of the following conditions is met:

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2.1.4 Evaluation of Bottom Connected Lines (DH)

For lines that cannot be eliminated from consideration based on the screening rules in Section 2.1.3, two avenues are provided for initial evaluation.

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2.1.5 Significant Temperature Threshold

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2.2 Screening and Evaluation Guideline

A methodology for assessing branch line susceptibility to thermal cycling and for determining thermal loading boundary conditions is summarized in this section and described in detail in Reference 7. The central element to the methodology is the thermal cycling application model, which has been developed from several extensive test programs [3, 5, 6].

2.2.1 General Overview

Thermal cycling in dead-ended branch lines occurs due to several physical phenomena, including the establishment and penetration of a well-defined flow structure in the branch line due to the primary reactor coolant line (RCL) header flow, thermal stratification, and interaction between the stratified fluid regions and the RCL-driven flow structure. The thermal cycling analytical model requires several input parameters such as the RCL velocity and temperature boundary conditions in addition to the geometry of the branch line piping configuration. The outputs of the analytical model provide an estimate of the thermal loading (boundary conditions) on the pipe, which is determined by the motion of the interface between hot and cold stratified fluid regions in the pipe. Assessment of thermal cycling susceptibility first consists of determining where cycling would theoretically occur based on the model, which may then be used to screen-in or screen-out the branch line based on geometric considerations, i.e., does the predicted thermal cycling location fall within a region of the branch line with a horizontal pipe run. If a branch line is determined to be susceptible, a thermal load definition is given that may be used for further (structural) analysis. The thermal load is defined by the fluid temperatures, heat transfer coefficients, and a periodic function (waveform) representing the motion of the thermal stratification interface.

Branch lines are generally attached to the top or bottom of RCS loop piping, but some lines intersect at the side or from other orientations. Thermal cycling and stratification phenomena are affected by piping orientation, such that branch line piping configurations may be broadly classified based on line geometry. This classification results in two generic configurations: the up-horizontal/horizontal (UH/H) configuration[1], which includes lines with up-horizontal (UH) geometry (Figure 2-3) and lines with strictly horizontal (H) geometry (Figure 2-4); and the down-horizontal (DH) configuration (Figure 2-5). Classification of attached piping is based on geometric considerations with the mechanisms for thermal cycling in these configurations fundamentally different. Common to each is the existence of a swirling vortical flow structure in the branch line induced by the high velocity flow in the RCS piping which decays with distance from the RCS. Thermal stratification and cycling in UH/H configurations is caused by the interaction of this swirl flow with the in-leakage of cold water from a high pressure source leaking through a normally closed valve between the source and the RCS. In contrast, valve in-leakage is not necessary in DH configurations, where thermal cycling occurs due to the cyclic penetration and retreat of the swirl flow in the branch line, combined with heat transfer to the environment that cools the lower horizontal line.

Application of the thermal cycling model first requires identification of the branch line based on the geometric configuration and potential for valve in-leakage. Branch lines with in-leakage potential and geometry that are above or at the same elevation as the primary reactor coolant loop piping fall within the UH/H model configuration. Branch lines with no in-leakage potential and geometry that are bottom connected to the RCS main loop piping fall within the DH model configuration. The UH/H and DH configuration application models are described in Section 2.2.2 and Section 2.2.3, respectively. Extensions of the model for use in more complex branch line configurations are described in Reference 7.

[1] Due to the similarity of the thermal cycling mechanisms, UH and H configurations will be considered together in Section 2.2 of this report. The UH/H notation will be used to identify model components that are applicable to both UH and H line configurations.

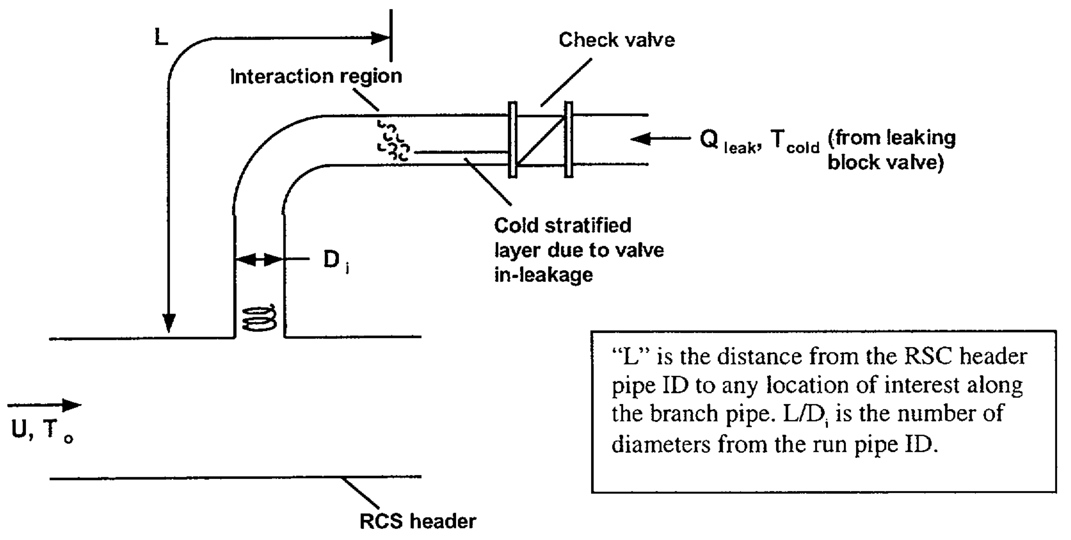

Figure 2-3
Illustration of UH Branch Line Piping Configuration

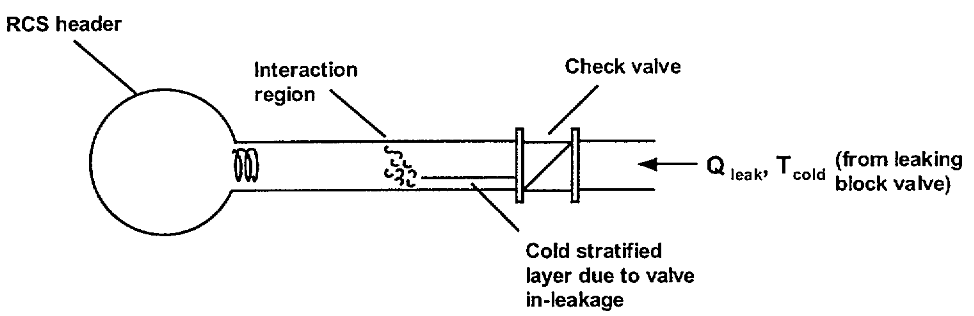

Figure 2-4
Illustration of H Branch Line Piping Configuration

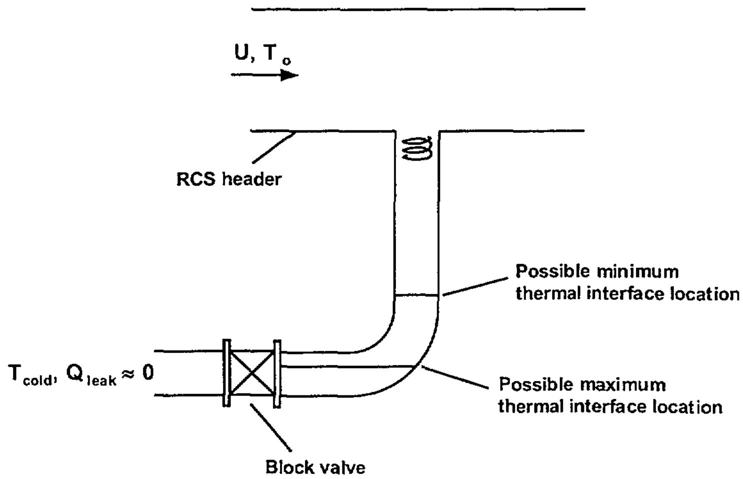

Figure 2-5
Illustration of DH Branch Line Piping Configuration

2.2.2 UH/H Configuration Thermal Cycling Model Overview

For piping configurations that fall within the UH/H classification, thermal cycling, and hence the thermal load, occurs due to the periodic axial motion of the cold stratified in-leakage layer interacting with the swirl penetration. Valve in-leakage establishes a cold stratified layer in a horizontal pipe run, which interacts with branch line swirl resulting in cyclic thermal loads applied to a region of the horizontal pipe segment. The average thermal cycling location ($\bar{x}_m$) is dependent on the branch line swirl strength, valve in-leakage flow rate and temperature, RCL temperature, and geometry. The associated thermal load is based on a simplification of the complex fluid dynamic structure in the thermal cycling interface region, as illustrated in Figure 2-6.

For thermal cycling screening and evaluation, the thermal cycling model provides estimates of several parameters that are defined in Figure 2-6[2]. These outputs are:

- the average cycling location, $\bar{x}_m$, measured from the RCS piping inside diameter, of the interaction region between hot and cold fluid regions,
- the axial and azimuthal variation of fluid temperatures in the hot and cold stratified fluid regions, T_1 and T_2,
- heat transfer coefficients associated with hot and cold fluid regions, h_1 and h_2,
- the height of the cold stratified layer, H, and
- the prescribed motion of the thermal stratification interface, $x_m(t)$.

[2] Note that circled numbers indicate fluid regions where the relevant thermodynamic and transport properties are approximately constant. The labels "State 1" and "State 2" indicate the extreme limits of the thermal cycling interface motion as a function of time.

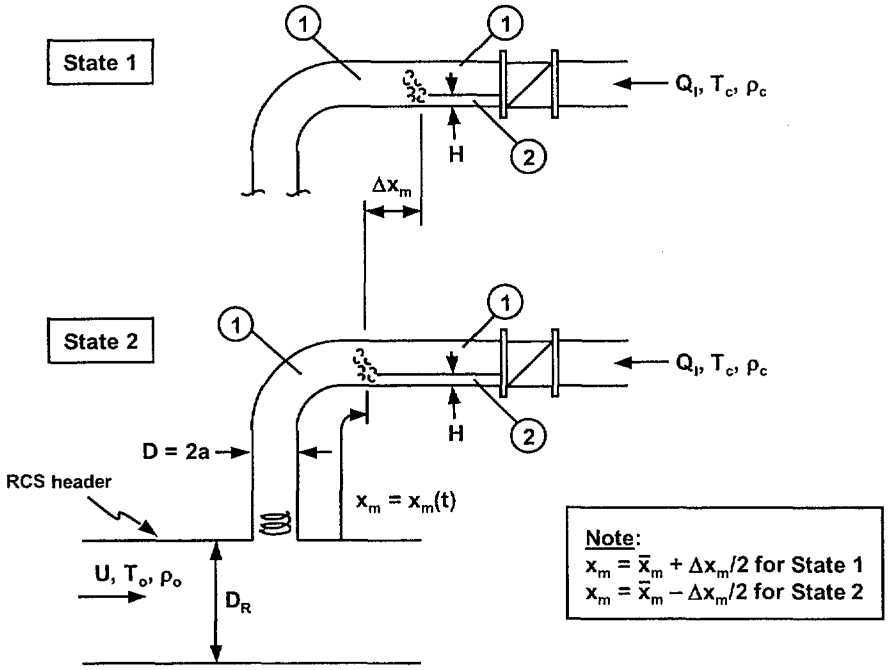

Figure 2-6
Thermal Cycling Model Parameter Definitions and Fluid Boundaries for UH/H Branch Line Configurations

An approach to evaluate UH/H piping configurations is outlined in Figure 2-7. Details are described in Reference 7.

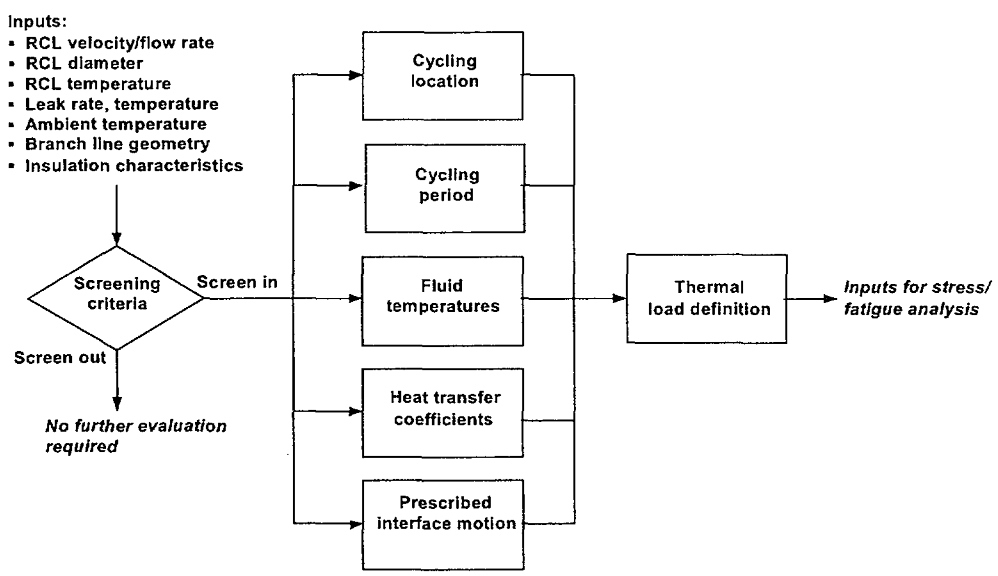

Figure 2-7
UH/H Thermal Cycling Model Evaluation Approach

2.2.3 DH Configuration Thermal Cycling Model

In this section, the thermal cycling screening and evaluation methodology for DH piping configurations is summarized. The general approach is similar to the UH/H methodology described previously. In the DH configuration, thermal cycling occurs due to cyclic penetration, break down, and retreat of a thermal stratification interface that is formed by the interaction between swirl penetration and fluid in the branch line. The branch line fluid has a lower temperature (higher density) than the RCL header due to heat transfer to the environment. Cyclic motion of the thermal interface provides a thermal load to the pipe, which may lead to thermal fatigue.

Unlike the UH/H methodology, valve in-leakage is not required for thermal cycling to occur in DH line configurations. Note that while cyclic valve out-leakage has been previously attributed to failure in one DH configuration plant leakage event, it is generally believed that the cyclic penetration and retreat of the thermal interface is a fundamental mechanism for thermal cycling in drain lines, residual heat removal suction lines, and similar lines. Valve leakage effects are not considered in the methodology for DH line configurations.

As with the UH/H configuration, the thermal load is determined based on the motion of the thermal interface separating two fluid regions: the swirl penetration region (1) and the cold-trapped region (2) (see Figure 2-8). The thermal cycling model provides estimates of the following parameters:

- the maximum penetration, x_m, of the thermal interface from the RCS piping inside diameter,
- the height of the hot stratified layer, H,
- the axial and azimuthal variation of fluid temperatures in the hot and cold stratified fluid regions, T_1 and T_2,
- heat transfer coefficients associated with hot and cold fluid regions, h_1 and h_2, and
- the prescribed motion of the thermal stratification interface, $x_m(t)$.

Note that the region (1') shown in Figure 2-8 identifies the fluid region in which the temperature increases as the thermal interface penetrates into the branch line and decreases after retreat of the thermal interface.

An approach to evaluate DH piping configurations is outlined in Figure 2-9 and detailed in Reference 7. The methodology described in this section is applicable to simple DH line configurations. Alternate and more complex line configurations can be evaluated using the methods described in Reference 7.

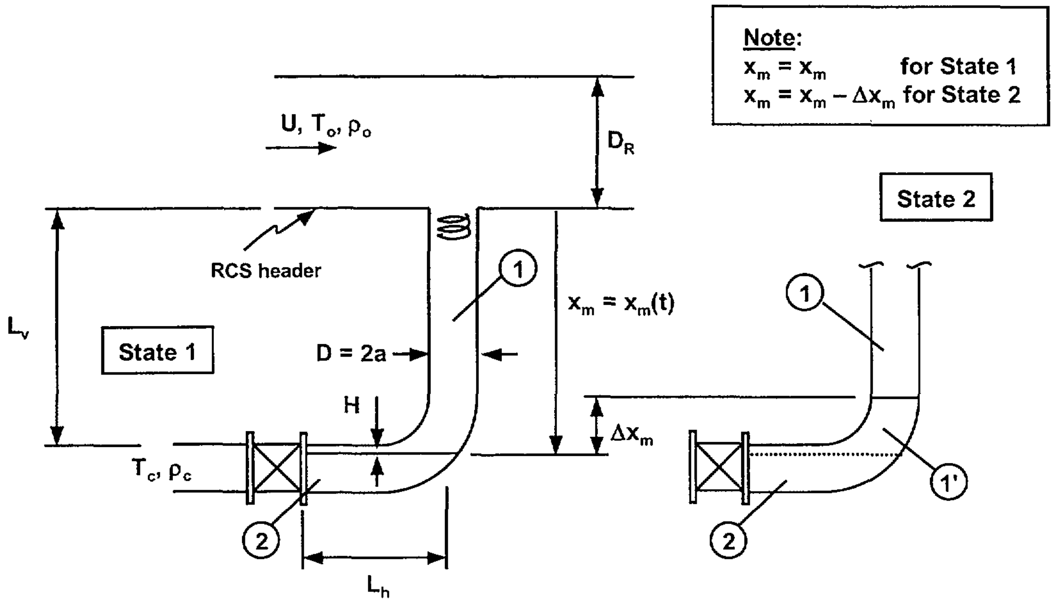

Figure 2-8
Thermal Cycling Model Parameter Definitions and Fluid Boundaries for DH Branch Line Configurations

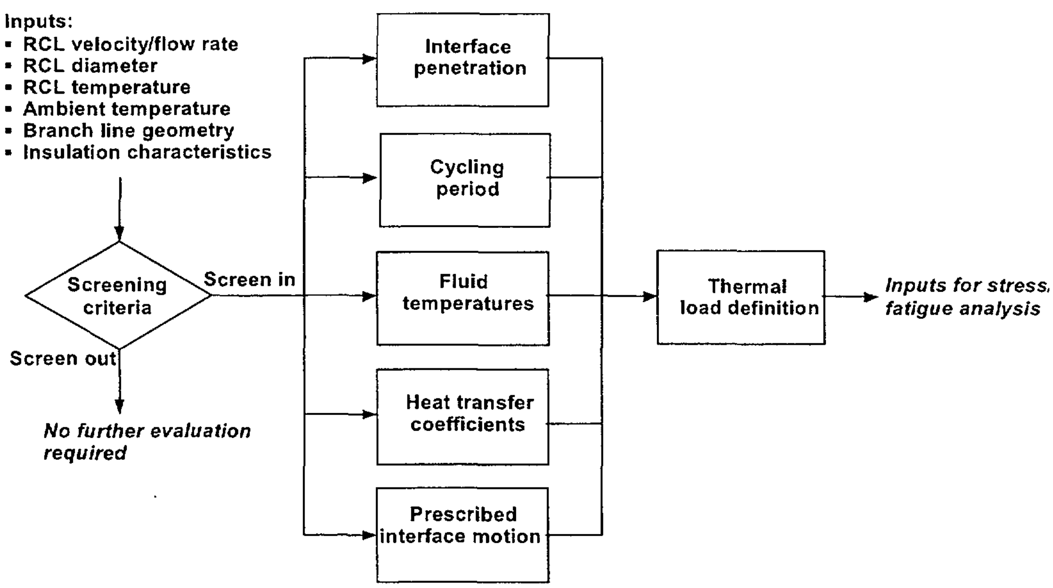

Figure 2-9
DH Thermal Cycling Model Evaluation Approach

2.2.4 Guide to Using MRP-132 to Perform Screening and Evaluation

Table 2-1 provides a convenient guide to the Sections of MRP-132 [Reference 7] that should be used in the performance of the various Screening and Evaluation tasks described in Section 2.1.

Table 2-1
Cross Reference of Screening and Evaluation Activities between this Guideline and MRP-132

Topic	MRP-132 Methodology Reference	MRP-132 Example Reference
2.1.1 Screening of Top or Side Connected Lines (UH or H)	Swirl penetration cannot reach the upper horizontal sections: §2.2.1 & Figure 2-6	Appendix D (see examples on p. D-1)
2.1.2.2 Thermal Cycling Not Significant Based on Modeling [UH/H]	• Interface not in the Horizontal: §2.2.2 • Determining ΔT: §2.2.6; Eqn 2.2.28 evaluated at x_1 corresponding to x= $\bar{x}_m + \Delta x_m/2 + D$	• See Example 4.1.1, noting commentary on p. 4-3. • See Example 4.1.2 to calculate ΔT loading, which should be compared to allowable threshold given in Sect. 2.1.5, herein.
2.1.2.3 Significant Thermal Cycling – Fatigue Analysis [Load Definition for UH/H]	§2.2; Figure 2-5.	See Example 4.1.3.
2.1.3 Screening of Bottom Connected Lines (DH)	Swirl penetration cannot reach the lower horizontal sections or swirl penetration causes the horizontal to be constantly heated: §2.3.1 & Figure 2-12	Appendix D (see examples on p. D-1)
2.1.4.2 Thermal Cycling Not Significant Based on Modeling [DH]	• Interface not in the Horizontal: §2.3.2 • Determining ΔT: §2.3.4; Eqn 2.3-25	• See Example 4.2.1, noting commentary on p. 4-13. • See Example 4.2.3 to calculate ΔT loading, which should be compared to allowable threshold given in Sect. 2.1.5, herein.
2.1.4.3 Significant Thermal Cycling – Fatigue Analysis [Load Definition for DH]	§2.3; Figure 2-11.	See Example 4.2.4 with cycling period determined from Example 4.2.2

2.3 Monitoring Guideline

For systems subject to in-leakage, parameters such as line pressure or valve leakage can be monitored to define leakage rate or to directly show that in-leakage does not occur. Temperature monitoring can also be used to show that cycling is not significant or that no leakage is occurring. An effective monitoring program that shows that valve in-leakage is not occurring can significantly reduce inspection requirements as indicated by the figures in Section 2.1.

If an analytical approach is being used, and if the resulting usage factor is high, implementation of monitoring has the potential of showing that the cycling is not as severe as predicted by the analysis.

Reference [11] provides a detailed assessment of monitoring, and may be used as a supplement to the guidelines included herein.

2.3.1 General Monitoring Criteria

Monitoring can consist of several different approaches. For systems subjected to in-leakage, the monitoring can quantify the in-leakage rate or confirm that it does not occur (by measuring pressure, valve leakage, etc.), or detect the thermal effects of in-leakage at or near the affected locations (by using temperature sensors). For drain lines, the monitoring must be targeted toward temperature sensing at the affected locations.

In-leakage occurs because a valve is not closing tightly enough to prevent leakage past the seat or due to disk deterioration or deformation. Since the amount of in-leakage could change with time (mainly as affected by open/close cycles), any monitoring to detect in-leakage or its effects must be ongoing and not discontinued after monitoring results indicate that in-leakage is not occurring. When using monitoring data either to show that thermal cycling is not significant or to supplement input for analysis, monitoring data must be taken and evaluated following each heatup from cold shutdown or after each open/close cycle of the normally closed valve that could result in the in-leakage.

For drain lines or other DH line configurations, monitoring need not be conducted during each operating cycle, since there is no source of in-leakage or other phenomena that will significantly change the conditions that will lead to thermal fatigue cycling. It is sufficient to take data during normal plant operation during one operating cycle. If there is no evidence of any significant thermal cycling as defined by 2.1.4.1, then the instrumentation may be removed after one operating cycle. If results from thermal monitoring are used to supplement an analytical evaluation of fatigue usage or flaw tolerance, the monitoring may be removed after two plant operating cycles, provided that results are consistent and conservative relative to that used in the analytical evaluation.

Note that the need for monitoring may need to be re-assessed if there are significant changes to RCS conditions (e.g., power up-rate).

If a monitoring operation commences after an evaluation is performed, the results need not be used to re-determine an inspection frequency until the next refueling outage.

2.3.2 Monitoring as an Inspection Alternative

Following the previously described assessments, locations may be identified as candidates for inspection or further examination. Installation of monitoring, as an alternative to future inspections, can be considered as described by this report. However, discovery of significant thermal cycling with the monitoring could indicate the potential for thermal fatigue or cracking.

Thus, inspection is recommended for locations if they are potentially susceptible to thermal cycling, as discussed 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.

2.3.3 Temperature Monitoring

Temperature monitoring can be accomplished by using sensors such as strap-on thermocouples or resistance temperature detectors. For monitored locations, it is recommended that the sensors be in contact with the surface 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 primary sensor be at the bottom of the pipe where cold leakage would exit the first check valve away from the RCS piping. Supplemental sensors should be placed at 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 in-leakage. Several top-bottom sensor sets should be placed in the range of Δx_m which for UH cases should be computed for the expected credible range of Q_l. (Δx_m and Q_l are defined in Figure 2-6.)
- For piping that exits the RCS piping horizontally, it is recommended that the sensors be placed at the check valve to detect in-leakage effects. Additional sensors could be placed on the line to detect cycling.
- 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 or to refine a thermal loading definition. Additional sensors could also 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 swirl 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 or similar DH lines.
- Steady stratification is typically seen in top or side connected lines that do not experience in-leakage and is the result of natural convection cells forming due to heat loss from the enclosed fluid. Even for piping geometries that are beyond the swirl 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.

For un-insulated drain lines, thermography may be utilized to show that conditions are steady and cold during normal operation. Any evidence that swirl penetration is contributing to the heating of the un-insulated piping should be further evaluated.

When temperature sensor data on piping systems are used in defining the predicted cyclic temperature ranges or number of cycles for evaluation of the affected locations, the applicability of the data must be qualified:

- Temperature sensor data taken on the outside of a pipe cannot be directly used for predicting inside fluid temperatures. There is a thermal time lag and a response attenuation that must be taken into consideration. The severity of the thermal load may not be apparent due to the pipe metal response to inside fluid temperature and axial and circumferential heat transfer effects. For high frequency loadings, it may be that no effects of fluid temperature fluctuation are observable on the outside of the piping.

The location of the temperature sensor may not be at the location that would experience the most significant cycling. In this case, guidance must be taken from the evaluation model to show that the observed data is corrected for the location effect.

2.3.4 Pressure Monitoring

Pressure monitoring may be used for lines potentially susceptible to in-leakage. The region between the normally closed block 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 normally closed block valves).

2.3.5 Valve Leakage Determination Guideline

The following is a discussion of methods that can be used to determine the rate of leakage through a leaking safety injection, or charging block valve. The discussion centers on the methods typically used in nuclear plants for assuring adequate leak-tightness of containment isolation valves. Alternate methods are acceptable.

2.3.5.1 Safety Injection Valve Leakage Testing

Figure 2-10 shows a typical configuration and valve lineup for the safety injection lines for Westinghouse-designed plants that have capability for use of the chemical and volume control (charging) pumps to perform a high-head safety injection function. For plants with in-leakage potential, there is a single (or two in parallel) remotely operated normally closed valve (and multiple check valves) between the charging pump header and the RCS. Containment isolation is typically assured by a check valve inside containment (shown as SI-3) and the remotely operated normally closed valve outside containment (shown as SI-1). There are generally a number of locally closable maintenance valves (e.g., SI-4). (The actual valve line up may vary from plant to plant.)

For the configuration shown in Figure 2-10, three types of tests could be used to quantify leakage past the single closed valve:

- Testing During Plant Operation – For testing during plant operation, plant operators would be required to measure leakage flow from equipment to a vent, drain or test connection shown just outside containment (SI-2). By opening the valve in this small line, the region of piping between the inboard containment check valve and the potentially leaking valve could be depressurized. By collecting the amount of leakage from this line (after drainage), the total leakage from the outboard closed valve (and through the inboard series of check valves SI-6, SI-5 and SI-3 – expected to be minimal) could be determined for the test condition. Several plants perform this testing using graduated cylinder beakers to collect the leakage. (Means of correcting the leakage so determined to plant operating conditions are discussed later.)
- Testing During Plant Shutdown – During plant shutdown conditions, containment penetration leakage testing is conducted. For this testing, the leak-tightness of both the inboard and outboard containment isolation valves is tested by pressurization of sections of the piping. Local maintenance valves (e.g., SI-4) may be closed to facilitate this testing. This same testing could be used to quantify the leak-tightness relative to valve in-leakage, with the assumption that the in-leakage and out-leakage characteristics would be the same. In Figure 2-10, the testing could be accomplished by closing the maintenance valve (SI-4) inboard of the containment check valve. The piping between the two block valves (SI-4 and SI-1) would then be pressurized, and the amount of leakage determined. The testing could be done with air, which is the normal method for penetration leakage testing, or with water. The resulting leakage rate would be the sum of the leakage past the two valves. (Note that if additional maintenance valves were present inside containment, the uncertainty due to the in-leakage path could be reduced.)
- Alternate Test Procedure During Shutdown – This would be a variation of the online testing approach described above. With this approach, the charging header would be isolated and pressurized with an alternate source of pressure. By testing with the RCS at near ambient conditions, the potential for back-leakage from the RCS could be minimized.

2.3.5.2 Alternate Charging In-Leakage Testing

Figure 2-11 shows a typical arrangement for the charging lines for Westinghouse and Combustion Engineering designed plants. Since all of the lines are inside containment, it is probably not possible to test the lines during operation as described for the safety injection lines above, but this approach will also be described.

- Testing During Normal Operation – With this approach, leakage rate samples could be collected from vent or drain lines in the isolable sections of piping between the flow control valves and the RCS check valves (e.g., A2, C3 or L3 shown in Figure 2-11). For this approach, the uncertainty due to check valve (A1) leakage for the auxiliary spray line would be larger since there is generally only a single check valve that would prevent leakage from the RCS.
- Integrated Testing During Shutdown – For this approach, the section between charging/auxiliary spray lines and the outboard containment isolation valves would be isolated (e.g., by closing A3, C4, L4 nearest the RCS and M5 and/or M6 outboard of the heat

exchanger). The isolated section would then be pressurized with water or air (e.g., through M1 or M2), and the integrated leakage would be determined. The disadvantage of this method is that the exact location of the leaking valve would not be known.

- Individual Valve Testing - A variation of the above procedure would be to collect leakage from the vent/drain lines downstream of the control valves (e.g., at valves A2, C3 and L3) while pressurizing the region upstream of the control valves (as above). This approach has the advantage of testing the individual valves (A3, C4 and L4) for leak-tightness. Any leakage attributable to the downstream check valves (A1, C1/C2 and L1/L2) could be determined prior to pressurizing the region upstream of the control valves.

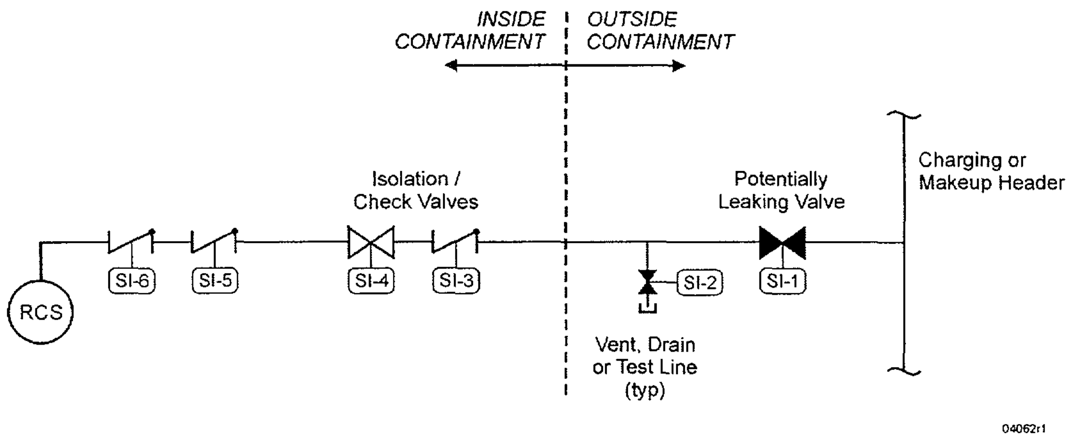

Figure 2-10
Typical High-head Safety Injection Line

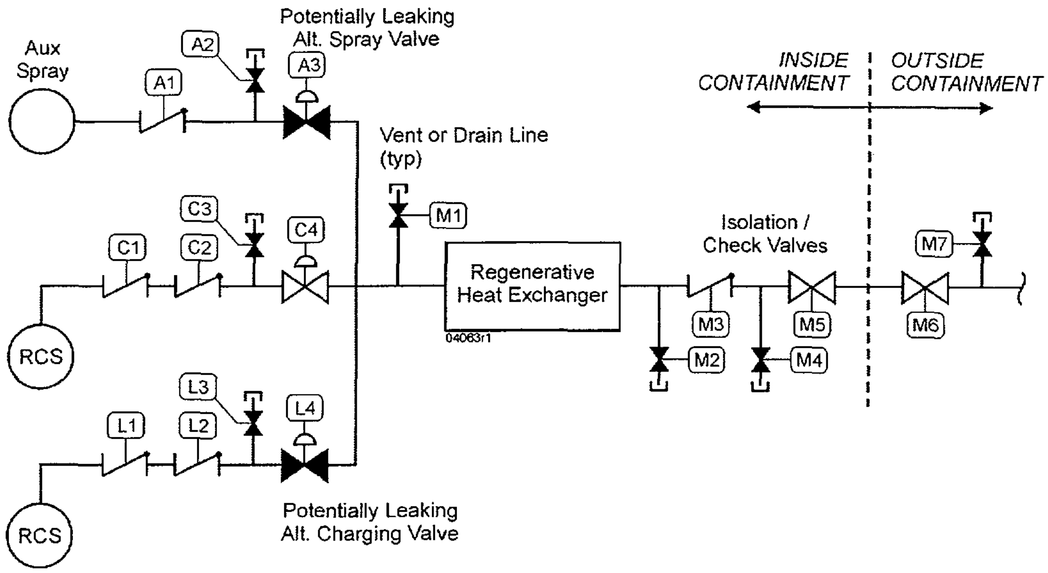

Figure 2-11
Typical Charging/Auxiliary Spray Lines

2.3.5.3 Correction to Normal Operating Conditions

The difference between test conditions and the conditions during normal plant operation must be considered in determining leakage rates used in the assessments of Section 2.1.

2.4 Inspection Guidelines

This section provides guidance for piping system inspection where required by the evaluations described in Section 2.0 and reflects EPRI studies documented in Reference 12. 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 Code, Section XI, IWB-3500. Radiographic methods do not typically detect cracking that is less than 10 percent of wall thickness [12].

Reference 12 contains a generic procedure for the ultrasonic examination of small-diameter piping that may be used. Visual or radiographic examinations may be performed using established utility procedures.

If thermal fatigue cracking or crazing is detected, actions should be undertaken to determine the reasons for the cracking (e.g., valve in-leakage) and to eliminate the source of the loading. Means of mitigating thermal fatigue loadings are discussed in Reference 10.

2.4.1 General Examination Requirements

UT examination of small (≤ 4-inch) diameter piping is more difficult than examination of larger diameter piping. Pipe wall curvature becomes a factor in back wall 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. However, it is possible to detect thermal fatigue cracking, although crack depth sizing is not sufficiently accurate to be reliable. Special UT transducers are required on small diameter piping, especially elbows. Refer to Reference 12.

In any regions where examinations indicate the potential existence of cracking or crazing, sufficient supplemental examinations shall be performed to determine if flaws exist.

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

For geometries that are not specifically described below, the same philosophy for examination of locations and volumes shall be used to develop line-specific inspection guidelines.

It is not necessary that 100 percent of the recommended examination volumes be inspected. Experience shows that high cycle thermal fatigue cracking due to swirl/turbulence penetration and valve in-leakage effects will be fairly wide spread. Thus, if full examination is not possible due to obstructions, weld crowns, etc., inspections that cover most of the examination volume is adequate to detect the presence of thermal fatigue cracking.

High-cycle thermal fatigue cracking is generally characterized by multiple initiation sites with the presence of crazing and fairly tight cracks, and may occur in either welds or base metal. In addition, inspection may be required in relatively small diameter piping. Thus, examiners must be properly trained to assure they can detect thermal fatigue cracking. The required training is as follows:

- Previous formal qualification for piping ultrasonic examination such as the ASME Code, Section XI Appendix VIII qualifications administered by the Performance Demonstration Initiative (PDI) or 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.

To meet this second requirement, a computer-based training module has been developed by EPRI 13].

2.4.2 Inspection Volumes

2.4.2.1 Inspection of Lines with Potential In-leakage (UH/H Lines)

For those lines that are potentially susceptible to in-leakage and might be affected by thermal fatigue, inspection requirements are outlined in Section 2.0. The potentially susceptible lines are those attached to either the top or side of the reactor coolant system piping.

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Figure 2-12
Examination Zones for Butt-Welded Lines Vertically Upward from RCS Piping

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Figure 2-13
Examination Zones for Socket-Welded Lines Vertically Upward from RCS Piping

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Figure 2-14
Examination Zones for Horizontal Lines from RCS Piping

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Figure 2-15
Examination Zones for Butt-Welded Bottom Connected Lines

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Figure 2-16
Examination Zones for Socket-Welded Drain Lines

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**Figure 2-17
Examination Volume for Thermal Fatigue Cracking in Piping Welds Less than NPS 4 [14]**

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**Figure 2-18
Examination Volume for Thermal Fatigue Cracking in Piping Welds NPS 4 or Larger [14]**

2.5 Thermal Fatigue Mitigation Guideline

Thermal fatigue mitigation is an approach that may be used to eliminate or reduce the potential for or severity of future thermal fatigue cycling. Mitigation should also be considered if inspection shows cracking or crazing that are due to thermal fatigue. Actions may include the following:

Preventative maintenance, such as valve maintenance to reduce in-leakage potential

Plant modifications, such as piping rerouting, valve relocation or addition of insulation

Changes in plant operations

Operation prior to mitigation must be taken into consideration in any analytical methods to show acceptance.

Reference 10 provides further information on methods that may be used to reduce thermal fatigue effects. Considerations are given to both stagnant and normally flowing lines attached to the reactor coolant piping. Section 2.6 of Reference 7 discusses parameters for which the most benefit may be achieved if physical modifications are considered.

2.6 Guidance for Performing Stratification Analysis

The following is a summary of the approach for evaluating the effects of thermal stratification and cycling on reactor coolant system (RCS) attached piping systems. The elements of the evaluation are the same as those that would be considered for new plant design in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, namely:

- Define the operating conditions, including loads (e.g., thermal expansion, seismic, dead weight, etc), steady and transient thermal boundary conditions, number of loading cycles, etc.
- Perform stress analysis to define primary stress intensities, and secondary and peak stress intensity ranges for all (or controlling) locations in the piping system.
- Use the stress ranges and defined cycles to determine CUF.
- Compare the values above with Code allowables.

ASME Code, Section XI, Appendix L presents a slightly modified approach, in that actual piping system loads and number of cycles can be used to perform a fatigue analysis.

2.6.1 Define Operating/Loading Conditions

The piping systems attached to the reactor coolant system are normally subject to the same steady state and transient loading conditions that affect the reactor coolant system. Thus, the first consideration is to define the reactor coolant piping system transients that affect the attached piping. The fluid conditions that each attached system experiences may be different from that of the RCS if the location to the first valve is quite long, or if the piping is bottom connected to the RCS.

Stratification/cycling and/or in-leakage from a high-pressure source may also affect the attached piping system, primarily during otherwise normal operating conditions. The transient thermal conditions for these loadings may be developed through methodology described herein, or modified using results from actual plant monitoring. Since the number of these loading conditions is expected to be related in some way to normal plant operating cycles, the relationship of the transient loadings to normal plant transients (e.g., number and severity per plant heatup/cooldown cycle) must be defined. Note that if there has been a change of operating conditions or if a change is projected, then this change should be considered in defining the loadings and number of cycles.

In accordance with ASME Code, Section XI, Appendix L, the evaluation can be conducted for the plant life or for a shorter period of time (e.g., up to the next inspection of the fatigue-critical location.) As stated, the plant loadings may be based on design conditions or actual plant conditions (e.g. temperature magnitudes or transient conditions).

2.6.2 Stress Analysis

Stress analysis for piping systems is typically conducted with specialized computer programs that perform an integrated evaluation of an entire system. This analysis can include:

- Thermal transient evaluation to determine the ASME Code transient temperature parameters (T_a, T_b, ΔT_1, and ΔT_2) for all loading conditions. (This analysis is sometimes conducted using separate computer programs and is directly input to the piping program.)
- Piping static and dynamic stress analysis for dead weight, seismic (primary and anchor movement effects) and other dynamic loads (if applicable). This results in piping moments for each loading condition in the piping system.
- Piping flexibility analysis to determine bending moments for all thermal loading conditions (including anchor movements).
- Primary stress intensity evaluation for all locations in the piping system.
- Load combination evaluation to determine the stress intensity ranges for all locations in the piping system.
- Fatigue analysis, and if required, thermal ratchet evaluation.

The Code piping evaluations as described above were developed for axisymmetric thermal loading conditions and generally did not consider steady or transient thermal stratification. Transient stratification produces three effects:

- Global bending effects in the piping system that produce modified piping thermal expansion moments in regions affected by the stratification as well as adjacent sections of piping,
- Local stresses at the region of stratification due to the non-linear stress distribution around the circumference of the pipe, possibly varying as the depth of stratification varies over time,
- Transient and/or steady through-wall stresses (currently defined by ΔT_1 and ΔT_2 in the Code). These stresses would also not be uniform around the pipe for stratified conditions.

The effects of stratification must include both the piping bending moments due to stratification and the additional local peak stresses due to a non-linear temperature difference across the pipe diameter.

The global bending effect of stratification must be evaluated by assessing the overall piping flexibility and thermal expansion movements as a result of the thermal stratification. Since the fluid and metal temperature do not necessarily vary linearly from the top to the bottom of the pipe, the equivalent moment-generating effect must be determined. To determine this loading, a local thermal stress analysis that considers the local fluid temperature profile and wall heat transfer coefficients must be conducted. From this analysis, the equivalent linear top-to-bottom temperature gradient can be determined for input into an overall piping system analysis program. Alternately, the moments generated by stratification can be input on individual piping segments to conduct the analysis. In the piping analysis, stresses due to global stratification bending effects are treated exactly the same as those due to thermal expansion.

In a stratified section, primarily axially oriented stresses develop even if the piping section is free to deform, due to the previously described nonlinear top-to-bottom temperature distribution. (A temperature distribution in any free body that is not linear in Cartesian coordinates will produce such a stress.) These stresses have the same characteristics as the more familiar non-linear through-wall stress distribution in that they will not result in gross thermal displacement of the piping system and they contribute only to fatigue. If a piping program is being used to perform stress analysis, this additional stress may be transformed into an equivalent ΔT_1 (choosing this term since the effects of any stress indices should be added to the resulting computed stresses).[3]

Conceptually, the fluid stratification in Figure 2-19 is analyzed to determine the temperature profile in the metal, as shown in the "T(r, θ)" curve in Figure 2-20. The equivalent moment generating linear temperature profile is the "V" curve in Figure 2-20. The nonlinear distribution is the last curve to the right of the "V" curve in Figure 2-20. The linear equivalent moment "V" temperature will result in moments that re-distribute according to the flexibility analysis, but are bounded by the two extremes in Figure 2-21.

The additional stresses due to through-wall thermal gradients should also be determined for each loading condition. Thermal transients produce relatively large ΔT_1 and ΔT_2 stresses, and T_a - T_b axial stresses that must be considered. However, there can also be through-wall temperature differences that occur with steady stratification that cannot be neglected.

2.6.3 Comparison to Code Allowables

Once these moments, piping thermal parameters and/or local stresses are determined, the piping fatigue analysis can proceed using the methods commonly used for piping analysis. The Code allowables for primary stress intensities should be unaffected by thermal loadings. The Service Level A/B stress intensity ranges must be evaluated in accordance with the NB-3650 of Section III of the Code. The cumulative usage factor (CUF) is compared to the allowable 1.0.

More detailed analysis may be conducted using the rules of the ASME Code, Section III, NB-3200, where the stresses at multiple locations around the circumference of a component may be considered.

[3] Discussions are underway in the ASME Code Committees to determine the appropriate treatment of the non-linear stratification temperature term. Consideration will be given to whether it is more appropriate to apply stress indices as if this were analogous to a ΔT_1 or a ΔT_2 type stress. In the interim, it is recommended that the analyst apply appropriate conservatism.

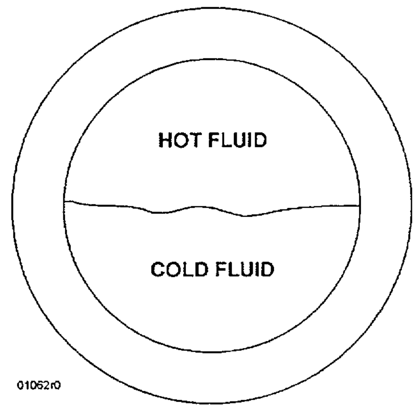

Figure 2-19
Stratification

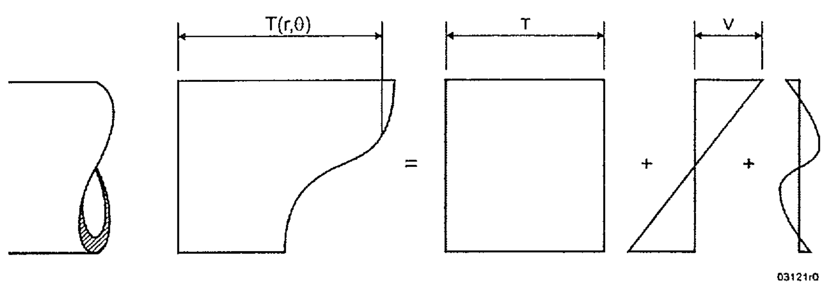

Figure 2-20
Stratification Temperatures

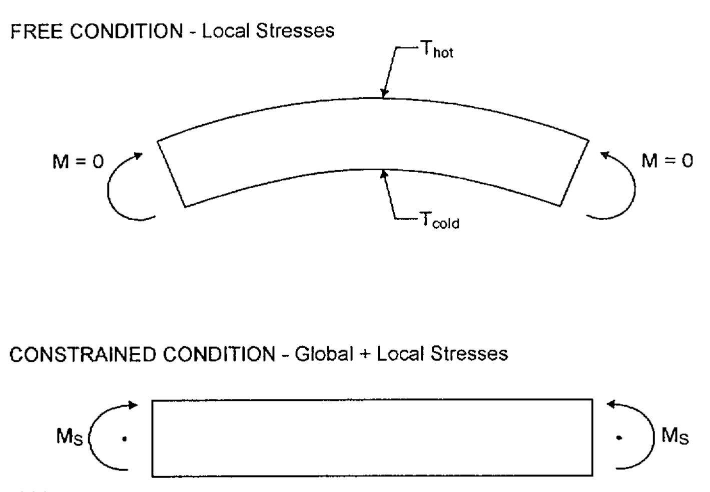

Figure 2-21
Bending and Moment Effects of Stratification

3 SUMMARY

This guideline has been prepared to meet the objectives and requirements of the NEI materials initiative [9]. It provides requirements classified as "needed" for evaluating normally stagnant non-isolable branch lines attached to the reactor coolant systems in PWR plants. This evaluation may require that certain locations be inspected to assure that through-wall cracking due to thermal fatigue does not occur in the future.

This guideline replaces an interim guideline published in 2001 [2] that recommended evaluation and potential inspection of safety injection lines and drain/excess letdown lines. The current guideline is based on more recent testing and analytical modeling, and may indicate that monitoring, evaluations, and/or inspections conducted in the past are not sufficient to assure that thermal fatigue cracking will not occur. Table 3-1 provides a comparison.

Table 3-1
Comparison of Interim and Final Guidelines

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Table 3-2 indicates the mandatory and needed action requirements of this document:

Table 3-2
Mandatory and Needed Requirements

Section	Requirement	Implementation Category [9]
1.3	An evaluation of each plant is required within two years of publication of this guideline. Any actions indicated by the evaluation shall be undertaken in a timely manner consistent with normal plant operation and refueling outages.	Needed
2.1	An assessment of non-isolable normally stagnant branch lines in the reactor coolant system is required. Flowcharts are provided that point to specific elements for screening, evaluation, inspection, and mitigation/repair/ replacement.	Needed
2.1.1/2.1.3	A screening approach is provided that may be used to eliminate the requirement for further evaluation/monitoring/inspection of qualifying lines. The use of this screening criterion is optional, provided that an evaluation in accordance with this guideline is undertaken.	Needed
2.1.2/2.1.4	These sections provide requirements for evaluation of affected branch lines, providing requirements for inspection locations and interval between inspections, based on results of evaluations and monitoring.	Needed
2.4.1/2.4.2	Requirements for Inspection are provided.	Needed

The remainder of the information that is provided in Section 2.0 provides guidance in performing the assessment noted above. Where referenced from the above sections, the information should be considered to be good practice [9] and used. Where alternate approaches are utilized, the alternate approaches shall be justified.

4 REFERENCES

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2. *Interim Thermal Fatigue Management Guideline (MRP-24)*, EPRI, Palo Alto, CA: 2001. 1000701.

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

4. 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.

5. *Materials Reliability Program, Identifying Thermal Cycling Mechanisms in Two Piping Configurations (MRP-54)*, EPRI, Palo Alto, CA: 2001. 1003081.

6. *Materials Reliability Program: Development of a Thermal Cycling Model for Un-Isolable Branch Line Piping Configurations (MRP-97)*, EPRI, Palo Alto, CA: 2003. 1003209.

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9. Guidance for Management of Materials Issues, Nuclear Energy Institute, Washington DC: May 2003. NEI, 03-08, and NEI Materials Guidelines Implementation Protocol, Revision 0, May 2004.

10. *Mitigation of Thermal Fatigue in Unisolable Piping Connected to PWR Reactor Coolant Systems (MRP-29)*, EPRI, Palo Alto, CA: 2000. 1001017.

11. *Thermal Fatigue Monitoring Guidelines (MRP-32)*, EPRI, Palo Alto, CA: 2001. 1001016.

12. *NDE Technology for Detection of Thermal Fatigue Damage in Piping (PWRMRP-23)*, EPRI, Palo Alto, CA: September 2000. 1000152.

13. *Computer-Based NDE Training for Thermal Fatigue Cracking (MRP-38),* EPRI, Palo Alto, CA: June 2001. 1001317.

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

A
THERMAL FATIGUE PROJECT REPORT SUMMARY

The MRP Thermal Fatigue project was composed of a series of tasks, all supporting development of a common industry approach to effectively manage thermal fatigue in non-isolable branch lines. The following provides summaries of the contents of the individual reports to support this program.

A-1 Guidelines

Interim Guidelines for Lines Where Leakage Previously Occurred

Interim Thermal Fatigue Management Guideline (MRP-24), EPRI, Palo Alto, CA: 2001. 1000701.

This guideline presents interim assessment and examination recommendations for Safety Injection and Drain/Excess Letdown lines. The purpose of this guideline was to provide utilities with a common approach to assess needs for additional actions for those lines where multiple occurrences of thermal fatigue leakage had occurred in the past. With this guidance, additional time would be available to develop the final guidelines presented in this report.

Non-Destructive Examination Method Development

NDE Technology for Detection of Thermal Fatigue Damage in Piping: PWRMPR-23. EPRI, Palo Alto, CA: September 2000. 1000152.

This report provides a state-of-the-art evaluation of NDE techniques for detection of thermal fatigue damage in small-diameter piping elbows and welds. In addition to providing the exploratory results for a number of candidate standard and advanced inspection techniques, recommendations are provided for using the techniques for detecting and sizing thermal fatigue cracking and crazing. A general inspection procedure is provided for ultrasonic examination of small diameter butt welds and base material for thermal fatigue damage. Recommendations are provided for personnel that are to conduct the examinations. Additional information is provided in Appendix C.

Computer-Based Training Module for Personnel Conducting Thermal Fatigue NDE

Computer-Based NDE Training for Thermal Fatigue Cracking (MRP-38), EPRI, Palo Alto, CA: June 2001. 1001317.

This computer-based training module was prepared to assist in qualifying NDE personnel to inspect for thermal fatigue in small-diameter piping systems. The differences between cracking due to intergranular stress corrosion cracking and fatigue cracking are emphasized. By studying the information on a computer, the student is able to simulate movement of an ultrasonic test sensor, and to observe the corresponding signal on a computer screen.

<u>Guidance for Installation of Monitoring Systems to Detect Thermal Fatigue Cycling</u>

Thermal Fatigue Monitoring Guidelines (MRP-32), EPRI, Palo Alto, CA: 2001. 1001016.

This report provides guidance to utility engineers for implementing monitoring programs on lines that could potentially experience thermal cycling due to valve in-leakage toward the RCS or due to swirl penetration into downward facing lines. The majority of the guidance relates to installation of thermal monitoring systems (e.g., thermocouples), but other types of monitoring are discussed. Monitoring for the purposes of detecting thermal cycling and for collecting data to support detailed thermal fatigue stress analysis are discussed. Data acquisition and transmission considerations are also provided.

<u>Mitigation of Thermal Fatigue Effects</u>

Mitigation of Thermal Fatigue in Unisolable Piping Connected to PWR Reactor Coolant Systems (MRP-29), EPRI, Palo Alto, CA: 2000. 1001017.

This report describes actions that can be taken if thermal cycling in unisolable lines is occurring. In addition, it addresses some operational considerations that can reduce normal transient severity in unisolable lines. It addresses plant modifications, operating procedures and maintenance activities that can reduce the potential for thermal fatigue cracking and leakage. The sources of thermal fatigue loading in piping systems and the susceptibility for thermal fatigue in various PWR piping systems are also discussed. Of key importance is the discussion on actions to prevent valve leakage into non-isolable Safety Injection system piping.

A-2 Supporting Studies

<u>TASCS Project (1989-1993)</u>

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

Following the occurrences of leakage in the Farley, Tihange and Genkai plants, and the issuance of NRC Bulleting 88-08, a research program was initiated to develop technology and practical tools that would allow utilities to evaluate thermal stratification, cycling and striping (TASCS) in affected lines. Laboratory tests were conducted to better understand stratified flow and the interaction between high-Reynolds-Number flow in reactor coolant piping and a normally stagnant branch line, with and without in-leakage flow toward the branch line. Results of other research and analytical projects were evaluated. Based on the testing, data correlation and fundamental analysis of stratified flows, a guideline was developed that included screening and evaluation correlations and tools to support TASCS-related evaluations. The methodology included in this report did not undergo NRC review, but informal feedback indicated that the methodology was not completely accepted by the NRC staff for analytical evaluation, primarily since it did not exactly predict the cycling location and loadings for the Farley Safety Injection line leak that occurred in 1987.

Laboratory Testing and Data Analysis

Materials Reliability Program, Identifying Thermal Cycling Mechanisms in Two Piping Configurations (MRP-54), EPRI, Palo Alto, CA: 2001. 1003081

Materials Reliability Program: Interim Report on Thermal Cycling Model Development for Representative Un-isolable Piping Configurations (MRP-81), EPRI, Palo Alto, CA, and U.S. Department of Energy, Washington DC: 2002. 1003527.

These reports describe a test program to identify and develop semi-empirical correlations for the physical thermal-hydraulic mechanisms that cause thermal cycling in normally stagnant dead-ended piping systems attached to reactor coolant system main loop piping. Three series of tests were performed. The first series of tests was conducted to evaluate the effects of cold-water in-leakage toward lines of the upper portion of main loop piping, such as those which caused the thermal fatigue-induced leakage for the safety injection lines at Farley and Tihange (1987-88). The second series of tests was conducted to assess the penetration of hot flow into downward running lines that led to thermal fatigue in the drain lines at TMI and Oconee (1995 and 2000). An additional set of tests was run in a high-Reynolds number facility to assess the penetration of swirl and turbulence from a high velocity flowing pipe into a stagnant attached line. These tests indicated that a "corkscrew" vortical flow structure was required for thermal cycling to occur. The critical dependent variables were reduced to several non-dimensional scaling curves to assist in development of methodologies for assessing thermal fatigue cycling.

Un-Isolable Line Evaluation Model Development (2002-2003)

Materials Reliability Program, Thermal Cycling Screening and Evaluation model for Un-Isolable Branch Line Piping Configurations (MRP-97), EPRI, Palo Alto, CA: 2003. 1003209This report describes the synthesis of the data from MRP-54 and MRP-81 to develop basic models for prediction of thermal cycling in normally stagnant branch lines. A model was developed to predict thermal cycling effects in up-horizontal and horizontal line configurations that might be affected by in-leakage of cold water toward a reactor coolant line with high-velocity high-temperature flow. This model predicted the cyclic interaction that would allow the location of the hot-to-cold water interface to cycle back on forth on the bottom of the horizontal line. A second model was developed to predict swirl penetration from the high-velocity reactor coolant line downward in a vertical line toward a colder horizontal line. These fundamental models were tested and benchmarked against several PWR plant configurations where thermal cycling had been observed to show that they could be used to conservatively predict thermal fatigue cycling.

Development of a Comprehensive Evaluation Approach

Materials Reliability Program, Thermal Cycling Screening and Evaluation Model for Normally Stagnant Non-Isolable Reactor Coolant Branch Line Piping with a Generic Application Assessment (MRP-132), EPRI, Palo Alto, CA: 2004. 1009552

This report builds upon the methodology described in MRP-97 and includes a more comprehensive evaluation model using results from the TASCS program. It includes screening criteria to define which geometries/systems would not be susceptible to thermal cycling as well as a methodology for determining the thermal loading in piping where cycling is predicted to occur. Examples of applying the methodology are included so that utility engineers can easily evaluate systems in their own plants that might be affected by thermal fatigue. Using plant data obtained during individual plant training, a generic plant assessment of the ranges of actual plant piping systems was conducted, reaching conclusions on the potential population of US PWR plants that might be affected by thermal cycling. A comparison of the methodology with the previous TASCS methodology is included as an appendix.

A-3 Other Documentation

Operating Experience

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

Materials Reliability Program: Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant Systems (MPR-85), Revision 1 to 1001006 (MRP-25). EPRI, Palo Alto, CA: 2003. 1007761.

This report and its revision describe the details, causes, thermal fatigue mechanisms and corrective actions taken for the significant worldwide thermal fatigue leakage events that have taken place in reactors similar to those in the US. Key information such as piping geometry, materials, crack orientation, temperature gradients and contributing causes are discussed. In addition, key results from plant monitoring for potential stratification and cycling are discussed. Based on this experience, the potential for thermal fatigue in potentially affected systems are described. The report revision expanded the scope to include several events that occurred in isolable sections of piping, but having some safety implications and being attributed to phenomena similar to that for the previous leakage events in the non-isolable sections. This document is an excellent resource for personnel not previously experienced with the implications of or causes for thermal fatigue in reactor systems. A summary of the relevant thermal fatigue leakage events is contained in Appendix B herein.

Thermal Fatigue Training

Lessons Learned From PWR Thermal Fatigue Management Training (MRP-83), EPRI, Palo Alto, CA: 2002. 1003666.

Starting in June 2000, thermal fatigue training on the Interim Thermal Fatigue Guidelines (MRP-24) was conducted at essentially every US nuclear plant site. As part of this training, an assessment of plant system susceptibility to thermal fatigue was conducted, based on the interim guidelines (for Safety Injection and Drain lines) and using engineering judgment for other systems. This document summarizes the observations from this training.

Computer Based Training Module

Thermal Fatigue Management for RCS-Attached Unisolable Piping – An EPRI/MRP Training Module (MRP-93), EPRI, Palo Alto, CA: 2003. 1007849.

The training notes from the thermal fatigue management training described in MRP-83 were posted to a computer-based training module. These notes, shown with additional annotation, may be used by plant personnel to gain an overall understanding of the thermal fatigue project and the Interim Thermal Fatigue Management Guideline for assessing RCS-attached, normally stagnant, non-isolable piping.

B
THERMAL FATIGUE / LEAKAGE EVENTS IN NON-ISOLABLE LINES

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

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

Biblis B

This event appears to be similar to the Obrigheim event and is associated with in-leakage of cold injection water from the Chemical and Volume Control system. The leak was 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 was not clear that the same type of lines exist in plants in the US and there had been no similar event here, no interim action was 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 was recommended.

Mihama 2

This leakage occurred in a normally 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 closed valve.

Oconee 1

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

References

1. *Materials Reliability Program: Operating Experience Regarding Thermal Fatigue of Unisolable Piping Connected to PWR Reactor Coolant Systems (MPR-85), Revision 1 to 1001006 (MRP-25).* EPRI, Palo Alto, CA: 2003. 1007761.

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	Through Wall Crack	
					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 (continued)
PWR Reactor Coolant Leakage in Non-Isolable Lines Attributed to Thermal Fatigue

Plant	Event Date	Initial Criticality Date	NSSS Vendor	Piping System	Through-wall Crack	
					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 (with numbers in parentheses indicating total for each type):
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)