ATTACHMENT

Braidwood Unit 1 Justification for the Deviation from MRP 2019-008, Technical Evaluation 635273

Reason for Evaluation/Scope:
This evaluation supports a deviation from the "needed" Interim Guidance in the Materials Reliability Program (MRP) 2019-008 document. This evaluation is prepared in accordance with ER-AA-4003.

The evaluation applies to the 1A Residual Heat Removal Suction line (1RC04AA) from the Reactor Coolant System Hot leg. The evaluation documents the acceptability of extending the inspection for the 1A RH suction line welds/pipe/elbow from the Spring of 2021 (A1R22) to the Fall of 2022 (A1R23).

Background:
Materials Reliability Program (MRP) 2019-008, "Interim Guidance for NEI 03-08 Needed Requirements for US PWR Plants for Management of Thermal Fatigue in Non-Isolable Reactor Coolant System Branch Lines", was issued April 1, 2019 (Reference 1). The requirements of MRP 2019-008 shall be implemented within two refueling outages after August 1, 2019.

The Braidwood review of MRP 2019-008 and required actions are documented in Action Tracking Assignment 4236453-03 (Reference 4). Based on the results of this review, required Non-Destructive Examinations were scheduled for A1R22 in the Spring of 2021 for Braidwood Unit 1. This schedule meets the MRP 2019-008 implementation requirement of "within two refueling outages after August 1, 2019". Interim Guidance 2 (IG2) from MRP 2019-008 applies to the Residual Heat Removal (RHR) pumps' suction lines from the Reactor Coolant System (RCS) Hot legs.

While reviewing the inspection reports in support of a Byron request and in preparation of sending the documentation to EPRI, it has been determined that the UT inspection for the 1A RH Suction line (1RC04AA) covered the 45-degree elbow and welds upstream of the branch line vertical drop from the A Reactor Coolant System (RCS) Hot leg (welds 1SI-02-47 and 1SI-02-48) and not the 90-degree elbow and welds upstream of the first horizontal piping run (welds 1SI-02-45 and 1SI-02-46). IR 4450126 has been initiated to documents this issue.

IR 4450129 has been written to add the MRP 2019-008 inspection for the 1A RH Suction line from the A RCS Hot Leg to A1R23.

This discrepancy is considered a "deviation" from the MRP 2019-008 "Needed" requirements.

The complete list of components that have been inspected in response to MRP 2019-008 in A1R22 (including the 1A RH train) is below:

- High Head Safety Injection Lines 1RC30AA, AB, AC and AD - Weld to the RCS nozzle and piping base metal downstream of the SI8900A-D check valves.
- RHR Suction Lines 1RC04AA - Piping Base metal and welds connected to the 45° elbow upstream of the first horizontal piping run from the RCS. This inspection does not meet the MRP 2019-008 requirements.
- RHR Suction Lines 1RC04AB - Piping Base metal and welds (Welds 1RC-11-3 and 1RC-11-4) connected to the 90° elbow upstream of the first horizontal piping run from the RCS.

* Note – The branch lines have an RC Equipment Part Number for the first section of the line off the RCS pipe.

The results of the inspections do not show any evidence of thermal fatigue (Reference 9).

For Braidwood Unit 2 the inspections are scheduled for A2R22, October 2021. This schedule meets the MRP 2019-008 implementation requirement of within two refueling outages after August 1, 2019.

Detailed Evaluation:
The operating experience (OE) from the MRP 2019-008 (Reference 1) is based on Westinghouse designed plants. Operating Experience reports document rejectable indications at the upstream elbow weld at the first elbow off the Residual Heat Removal (RHR) suction line. Subsequent thermocouple measurements indicated thermal cycling in the horizontal piping downstream of the elbow. This line had screened out of MRP-146 (Reference 2) because the MRP-170 (Reference 3) analytical software predicted the horizontal portion of the piping stays hot (no stratification/destratification cycling expected).

Exelon will implement the MRP 2019-008 "Needed" action to inspect the 1A RH Suction line for Braidwood Unit 1 in the Fall of 2022.

This extension does not result in negative consequences because:

1) Braidwood Technical Specifications Surveillance Requirement 3.4.13.1 requires that the RCS operational leakage be within limits by performance of RCS water inventory balance. In accordance with the Braidwood Surveillance Frequency Control Program, Surveillance Requirement 3.4.13.1 is required to be performed every 72 hours. The Braidwood U-1 RCS leakage rate surveillance (Reference 5) is conservatively performed on a daily basis (Reference 6).

   Although the Technical Specification limit for unidentified leak rate (ULR) is 1 gpm (Technical Specifications 3.4.13), the leak rate surveillance (Reference 5) includes action levels if the leak rate increases above baseline by a set factor. The current (As of 10/8/2021) baseline value for the Braidwood U-1 ULR is 0.021 gpm. This value is significantly lower than the 1 gpm limit. The Reference 5 leak rate procedure directs actions to identify the source of leakage. The capability to detect small RCS leakage allows taking actions before the potential degradation progresses to the point of challenging the structural integrity of the affected piping. This capability reduces the overall risk to the plant.

2) The welds connected to the 90-degree elbow on the 1A RHR suction line, 1SI-02-45 and 1SI-02-46, are within the scope of the Braidwood In-Service Inspection (ISI) Program. Weld 1SI-02-45 has been inspected in A1R22 (Spring 2021) and weld 1SI-02-46 has been inspected in A1R17 (Fall 2013). Neither exams documented any recordable indication (Reference 8A and 8B). These examinations could not be credited for MRP 2019-008 IG 2 requirements because they did not include the elbow base metal as shown in Figure 1B of MRP 2019-008. Since the OE that was part of MRP 2019-008 Interim Guidance 2 identified flaws in the welds only (Reference 9), the ISI exams results provide confidence that Braidwood Unit 1 is not incurring similar degradation.

3) The need for the MRP 2019-008 exams has been identified to determine if thermal fatigue indications are present in the stagnant piping due to the thermal cycling resulting from the proximity to the RCS Hot leg. The physical configuration between the 1A and 1B RH suction piping from the RCS are very similar (Reference 11 and 12). The fact that the MRP 2019-008 exam was not done on the 1A RH train welds does not raise concerns about thermal fatigue because the corresponding welds on the 1B RH train were inspected with no issues identified.

Conclusion:

The objective of the MRP 2019-008 interim guidance is to minimize the probability of thermal fatigue cracks that exceed allowable flaw dimensions in stagnant lines that are connected to the RCS. As discussed above, deferring the examination of the 1A RH Suction line from the 1A RCS Hot Leg by one cycle to A1R23 is not expected to pose significant risk to the integrity of the Reactor Coolant System for Braidwood Unit 1. MRP EPRI Programs owners have indicated that no flaws have been reported by nuclear plants in the United States that have performed these inspections and reported them, as required by MRP 2019-008 (Reference 13). Therefore, the deferral is acceptable.

References:

1. MRP 2019-008 "Interim Guidance for NEI 03-08 Needed Requirements for US PWR Plants for Management of Thermal Fatigue in Non-Isolable Reactor Coolant System Branch Lines", Electric Power Research Institute, April 2019 (attached below)
2. MRP-146, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines", Revision 2, Electric Power Research Institute, September 2016
3. MRP-170 "Thermal Fatigue Evaluation per MRP-146, Version 1.0, "Electric Power Research Institute," April 2006
4. Exelon CAP Action Tracking Item 04236453-03
5. Braidwood Surveillance procedure 1BwOSR 3.4.13.1, "Unit One Reactor Coolant System Water Inventory Balance Surveillance", Revision 40
6. Braidwood Surveillance Procedure 1BwOSR 0.1-1,2,3, "Unit One Modes 1,2 and 3 Shiftly and Daily Operating Surveillance", Revision 98
7. Braidwood Surveillance Frequency Control Program Revision 30
8. Braidwood Station A1R22 ISI Non-Destructive Examination Reports for 1A RH Suction Line:
    a) A1R22-UT-014 (1RC04AA-12, Weld 1SI-02-45)
    b) A1R17-UT-023 (1RC04AA-12, Weld 1SI-02-46)
9. Braidwood Station A1R22 MRP 2019-008 Non-Destructive Examinations Reports: (Pdf files of these reports are embedded in the EC Scope section of EC 635273)
    a) 2021-UT-032 (1RC30AA-1.5)
    b) 2021-VT-004 (1RC30AA-1.5)
    c) 2021-UT-033 (1RC30AB-1.5)
    d) 2021-VT-005 (1RC30AB-1.5)
    e) 2021-UT-034 (1RC30AC-1.5)
    f) 2021-VT-006 (1RC30AC-1.5)
    g) 2021-UT-035 (1RC30AD-1.5)
    h) 2021-VT-007 (1RC30AD-1.5)
    i) 2021-VT-008 (1RC04AA-12)
    j) 2021-VT-011 (1RC04AA-12)
    k) 2021-VT-009 (1RC04AB-12)
    l) 2021-VT-010 (1RC04AB-12)
    m) A1R22-UT-032 (1RC04AB-12)
    n) A1R22-UT-033 (1RC04AB-12)
    o) A1R22-UT-025 (1RC04AA-12)
    p) A1R22-UT-031 (1RC04AA-12)
10. ML17338A131, "Diablo Canyon Power Plant, Units 1 and 2 – Relief Request REP-RHR-SWOL, Request for Approval of Alternate for Application for Full Structural Weld Overlay"
11. Isometric Drawing 1SI-02, Revision F
12. Isometric Drawing 1RC-11, Revision E
13. E-Mail with Exelon MRP Owner, October 6, 2021

MRP Owner Contact.pdf

Embedded Attachment:

MRP 2019-008:

MRP 2019-008.pdf

Attachment 13 - E-Mail with Exelon MRP Owner, October 6, 2021

RE: MRP 2019-008 Deviation EC - Braidwood U-1

TP

Tamburro, Peter:(Exelon Nuclear)
To Panici, Giovanni:(Exelon Nuclear)

You replied to this message on 10/6/2021 2:24 PM.

Hey John

This statement is based on telephone conversations I had with the MRP-146 EPRI program owners Bob McGill and Chris Wax.

They basically said that no utility has reported issues for MRP 2019-008 inspection of there RHR elbows.

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MRP Materials Reliability Program **MRP 2019-008**

DATE: April 1, 2019

TO: PMMP Members
MRP Research Integration Committee (RIC) Members
MRP Pressure Boundary and Fatigue TAC Members

FROM: David Czufin, TVA, PMMP EC Chair
Brian Burgos, EPRI, MRP Program Manager

SUBJECT: Interim Guidance for NEI 03-08 Needed Requirements for US PWR Plants for Management of Thermal Fatigue in Non-Isolable Reactor Coolant System Branch Lines

ENCLOSURE:

1. Notification of Recent Thermal Fatigue OE Regarding Cross-Flow In-Leakage, December 15, 2017 (MRP 2017-026).

The purpose of this letter is to transmit NEI 03-08 "Needed" Interim guidance regarding thermal fatigue cracking.

NEI 03-08 Needed Interim Guidance for Management of Thermal Fatigue

Note: The requirements of this guidance shall be implemented within two refueling outages after August 1, 2019. Results of the exams shall be reported to the EPRI MRP within six months after completion.

IG#	Interim Guidance	Basis for Interim Guidance
1	DH lines previously exempted by the Generic Analysis option (i.e. R-Strat) described in MRP-146, paragraph 2.1.5.4 shall be inspected every other RFO if all the following conditions are true: a) cracking has been identified b) the cause of the cracking could not be identified and mitigated / eliminated.	The MRP 2015-019 (and subsequently, MRP-146 Rev. 2 Section 2.1.5.4) requires that sites with DH lines evaluated using the MRP-146, Appendix A generic evaluation complete a one-time examination and report the results to MRP. If cracking is then identified on these lines and the cause evaluation is unable to identify and mitigate the cause, then it should be conservatively assumed that a degradation mechanism is present. Therefore, a routine exam is justified consistent with other MRP-146 screened-in DH lines.
2	For those large bore (> 4") DH lines that previously screened-out as "HOT" (per MRP-146) a one-time examination of the piping at the first 45-degree (or 90-degree) elbow is	During a 2016 risk-informed in-service inspection of a 14" RHR Suction line (Class 1 piping), a weld flaw extending 24% of through-wall was identified by ultrasonic testing. This flaw was located in the Heat Affected Zone of the pipe to 45-

IG#	Interim Guidance	Basis for Interim Guidance
	required (as represented on Figure 1A). As a minimum, the inspection volume shall include the base metal and welds represented in Figure 1B. Note: credit may be taken for previous exam if: ▪ Previous exam volume requirements bound those of this IG	degree elbow weld on the upstream end of the elbow. This pipe line previously was exempt from examination since it was screened out as "HOT" per MRP-146 guidelines and MRP-170 software. Using an array of thermocouples at the flaw location and at additional downstream locations, high-frequency temperature oscillations were recorded at 100% power operations. An extent-of-condition examination on the sister unit in Spring 2017 revealed a similar (though less extensive) crack depth in a similar location. Sample RHR Hot Leg Suction Line - See Figure 1
3	Sites shall review MRP-146 "screened" out UH/H lines to determine susceptibility to in-leakage from cross-flow. To perform this determination, interconnected lines, or lines sharing a common header, with only check-valve isolation between RCS loops, shall be "screened-in" as potentially susceptible to in-leakage / cross-flow. For those new "screened-in" lines a one-time inspection shall be performed using the volumetric requirements of MRP-146 Rev. 2 (Figure 2-11 through Figure 2-14, as applicable) and as amended by IG #4 below. Note: credit may be taken for previous exam if: ▪ Previous exam volume requirements bound those of this IG	From MRP-146 Rev. 2, Section 2.1.1, it is possible that a UH/H branch line previously screened out of MRP-146 because no high pressure source (> RCS pressure) is present; however, such lines could still be susceptible to in-leakage from cross-flow if the following two conditions exist: a) The Loop branch lines are interconnected via a common header, AND b) A differential pressure exists between the interconnected RCS loops. Reference Enclosure 1- MRP letter 2017-026 for more background.
4	Future fatigue examinations of the bottom inner third thickness of base metal as indicated in Figure 2-20 of MRP-146, Rev. 2 shall be 1" wide. See Figure 2 below.	For the Boron Injection leakage event at a 4-loop Westinghouse-designed plant in 2017, the Metallurgical Examination showed cracking extending beyond the 1/2" width bottom dead center, as indicated in Figure 2 below. Therefore, TFFG recommends an increase to 1" width.

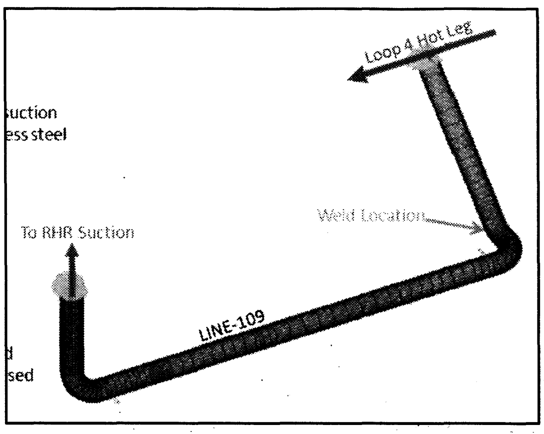
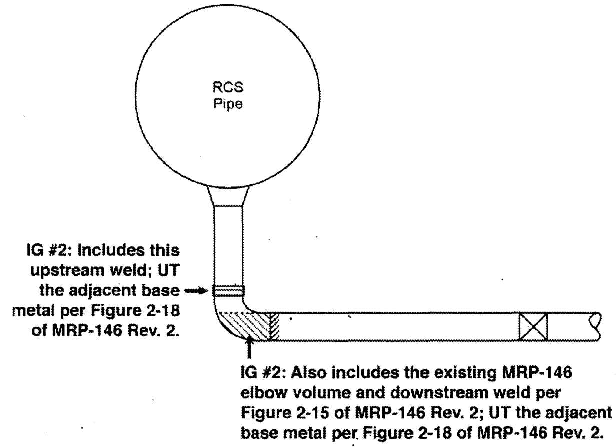


Figure 1A: I.G. #2 Sample Hot Leg RHR Suction Line

Figure 1B: I.G. #2 Sample Hot Leg RHR Suction Line



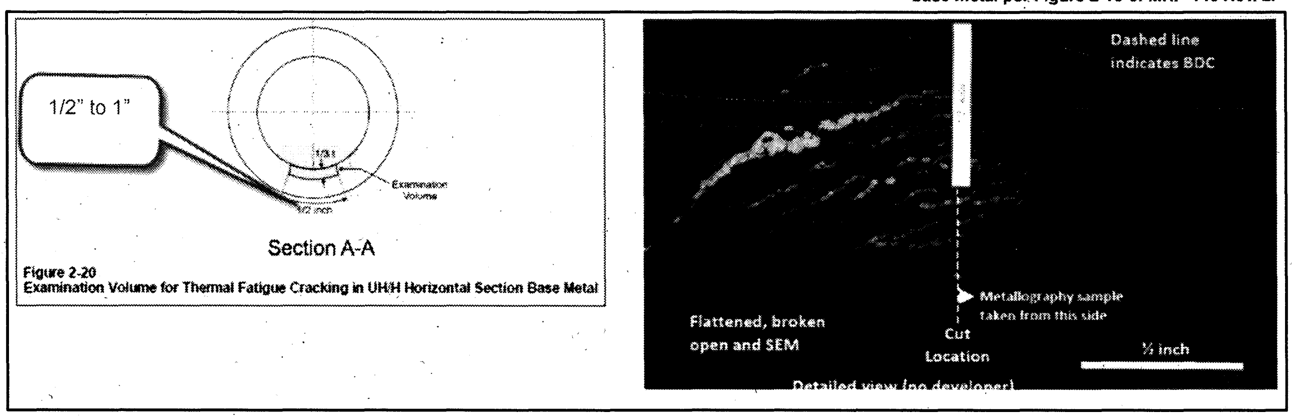


Figure 2: IG #4 Examination Volume Width Increase to 1 Inch

Background

Management of fatigue resulting from the interaction of thermally stratified fluids in reactor coolant system piping was first recognized in operating experience during the 1980s. Industry response to management of this fatigue mechanism was based on a combination of testing, analysis and operating experience (OE) assessments. This work resulted in development of thermal fatigue management guidelines including [2] & [3].

In response to numerous OE during the period 2013-2015, the MRP established a Thermal Fatigue Focus Group (TFFG) in 2015 to evaluate OE and determine if interim guidance was needed. MRP subsequently issued interim guidance via [4] and the portions of that guidance applicable to MRP-146 have since been incorporated into MRP-146 [2] in 2016.

In 2017 MRP issued letter MRP 2017-26 [1], included as Enclosure 1, which informed PMMP and MRP RIC members of Recent Thermal fatigue OE Regarding Cross-Flow In-Leakage. Whereas, cross-flow in-leakage is not new, being described in Section 1.0 of [5], Section 2.1.5 of [8], and Section 3.4.2 [10], it was thought to only be limited to periods of heat up/cooldowns when less than full number of reactor coolant pumps was running; thus, cross-flow in-leakage was not expected during normal operation. The MRP letter also indicated the TFFG was evaluating additional OE (2016-2017) to determine if interim guidance was warranted.

Consequently, this Interim Guidance letter communicates those findings. This added guidance is designed to minimize the probability of thermal fatigue cracks that exceed allowable flaw dimensions or result in forced or extended outages.

In parallel with this guidance, MRP Research Focus Area RFA 5.1 is seeking to improve MRP guidance using analytical CFD and Heat transfer analysis models validated against recent laboratory testing. Therefore, the TFFG is also tasked with identifying the research necessary to understand and effectively mitigate or manage the factors underlying the recent events. The recommended research is prioritized and managed by the appropriate Technical Advisory Committee (TAC) and Integration Committee.

Operating Experience Evaluation

Of the OE events summarized below, the TFFG has determined that four of these (highlighted) warranted additional interim guidance.

OE #	NSSS	System	Location	%TW	Inspection Basis	Config	IG Req'd ?	NEI 03-08 Needed Interim Guidance Reference and Bases (Enclosure 1)
1	W3	RCS Drain	Weld D/S (Elbow to Pipe)	58	EOC from U1 (2014)	DH	YES	IG #1
2	W4	Charging	3" Pipe to Nozzle	44	MRP-146	UH	NO	N/A Met Lab results limited, but inconclusive w.r.t. Thermal Fatigue as cause.
3	W4	RHR HL Suction	14" Pipe to Elbow	34	Sec XI	DH	YES	IG#2
4	W4	SI	Boron injection line 5D bend near RCS	leak	N/A	H	YES	IG #3 and 4
5	W4	RHR HL Suction	14" Pipe to Elbow	14	EOC from DC2	DH	YES	IG #2

<u>Other Observations for Consideration</u>

As described above, each of the recent OE events was analyzed by the Thermal Fatigue Focus Group (TFFG) to identify actions to prevent thermal fatigue cracking from exceeding allowable limits or resulting in forced or extended outages. In addition to the IG, the following observations are provided for consideration by engineers responsible for management of thermal fatigue:

A. Technical guidance and recommendations for proper and cost-effective maintenance of check valves is provided in NMAC Application Guide for Check Valves, NP-5479 [6] and EPRI Check Valve Maintenance Guide (Revision to TR-100857) [7].
   - In one of the OE events back-leakage through check valves combined with loop differential pressure during normal operations caused cross-flow in-leakage as described in Reference [1].
   - In another event, the apparent cause evaluation indicated a lift check valve (charging bypass line around isolation valve) was susceptible to leak by.

   Either of these leakage scenarios presents the potential for thermal fatigue cracking in the branch piping connecting to the RCS. Program owners are encouraged to take advantage of these resources.

B. Both units of another plant experienced rejectable weld indications at the upstream elbow weld at the first elbow off the RHR Suction Line. Subsequent thermocouple measurements have indicated thermal cycling in the horizontal piping downstream of the elbow. However, this line had screened out of MRP-146 because the MRP-170 analytical software predicted the horizontal portion of the piping stays hot (no stratification/destratification cycling expected). MRP Research Focus Area 5.1.6 is performing additional research to establish improved guidance on RCS stagnant branch lines addressing this particular RHR suction line configuration.

C. Use of Integrated Fatigue Management guidance as identified in MRP-148 [9] and the Fatigue Management Handbook, MRP-235 [10] are valuable tools to help manage fatigue degradation at operating units. In addition, Fatigue Management Handbook training materials are accessible to EPRI members via the training section of the EPRI MRP Cockpit. Program owners are encouraged to take advantage of these resources.

This information will be included in a future revision of MRP-146. If there are questions, please contact Paul Crooker, (650-855-2028, pcrooker@epri.com), or the undersigned.

David Czufin, TVA
PMMP EC Chair

Brian Burgos, EPRI
MRP Program Manager

REFERENCES

1. MRP Letter, MRP 2017-026, "Notification of Recent Thermal Fatigue OE Regarding Cross-Flow In-Leakage", December 15, 2017.
2. Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 2, 3002007853), Final Report, September 2016
3. Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines – Supplemental Guidance (MRP-146S, 1018330, January 2009
4. MRP Letter, MRP 2015-019, "Implementation of NEI 03-08 Needed and Good Practice Interim Guidance Requirements for Management of Thermal Fatigue", May 28, 2015.
5. EPRI Report TR-103581, "Thermal Stratification, Cycling and Striping", March 1994.
6. NMAC Application Guide for Check Valves in Nuclear Power Plants, EPRI NP-5479s, Rev. 1
7. EPRI Check Valve Maintenance Guide (Revision to TR-100857), October 2015
8. Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems, EPRI MRP-85, Rev. 1, October 2014.
9. Materials Reliability Program: Integrated Fatigue Management Guideline (MRP-148, Revision 1, 1025159, November 2012).
10. Materials Reliability Program: Fatigue Management Handbook (MRP-235, Revision 2, 3002005510, December 2015).

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MRP Materials Reliability Program MRP 2017-026

DATE: December 15, 2017

TO: PMMP Members
MRP IC Members

FROM: Terry Childress, Duke Energy Corporation, TS TAC Chair
Elliot J. Long, EPRI, Materials Reliability Program

SUBJECT: Notification of Recent Thermal Fatigue OE Regarding Cross-Flow In-Leakage

REFERENCES:

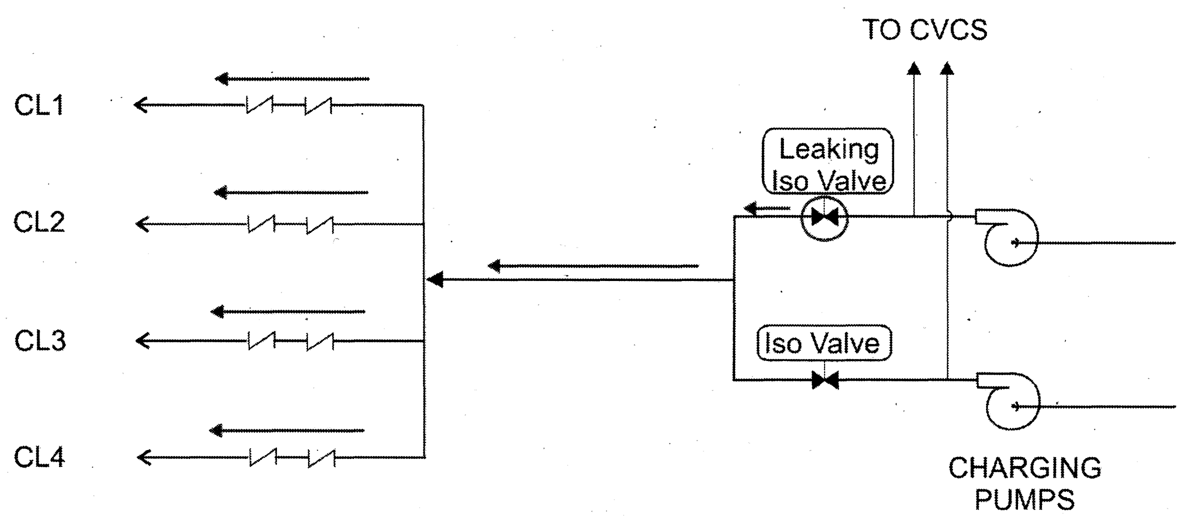
1. *Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146, Revision 2)*. EPRI, Palo Alto, CA: 2016. 3002007853.
2. NEI 03-08, Rev. 3, Guideline for the Management of Materials Issues, Nuclear Energy Institute, Washington DC: February 2017.
3. *Materials Reliability Program: Operating Experience Regarding Thermal Fatigue of Piping Connected to PWR Reactor Coolant Systems (MRP-85, Revision 1)*. EPRI, Palo Alto, CA: 2014. 3002003080.
4. *Materials Reliability Program: Fatigue Management Handbook (MRP-235, Revision 2)*. EPRI, Palo Alto, CA: 2015. 3002005510.

The MRP Technical Support Advisory Committee is issuing this notification of recent operating experience (OE) in piping and components potentially exposed to thermal fatigue. This experience has revealed a source of in-leakage from cross-flow that has resulted in through-wall thermal fatigue cracking and a forced plant outage. This cross-flow in-leakage is mentioned, but not specifically addressed, in current EPRI Materials Reliability Program (MRP) guidance. Therefore, this OE is being provided to you for your awareness and consideration at your sites. Please ensure the appropriate site management at your plants are made aware of this OE.

There are no new Materials Initiative (NEI 03-08) requirements associated with this communication; such requirements may be issued later after relevant OE is collected and investigated and the MRP Thermal Fatigue Focus Group (TFFG) has further dialogue on the matter.

BACKGROUND

MRP-146 [1] provides guidance and identifies “Needed” NEI 03-08 [2] requirements for the management of thermal fatigue in stagnant, non-isolable reactor coolant system (RCS) branch lines. For branch lines attached to the tops or sides of RCS piping, in-leakage (unintended fluid movement toward the RCS main loop) may interact with hot swirl penetration (generated from the RCS high speed flow) and result in thermal fatigue cycling. Typically, this in-leakage results from a leaking isolation valve that allows fluid from the charging system (high pressure source) to enter the stagnant safety injection system during normal operation (see Figure 1).



Note: Some piping configurations only have single check valve isolation from the RCS.

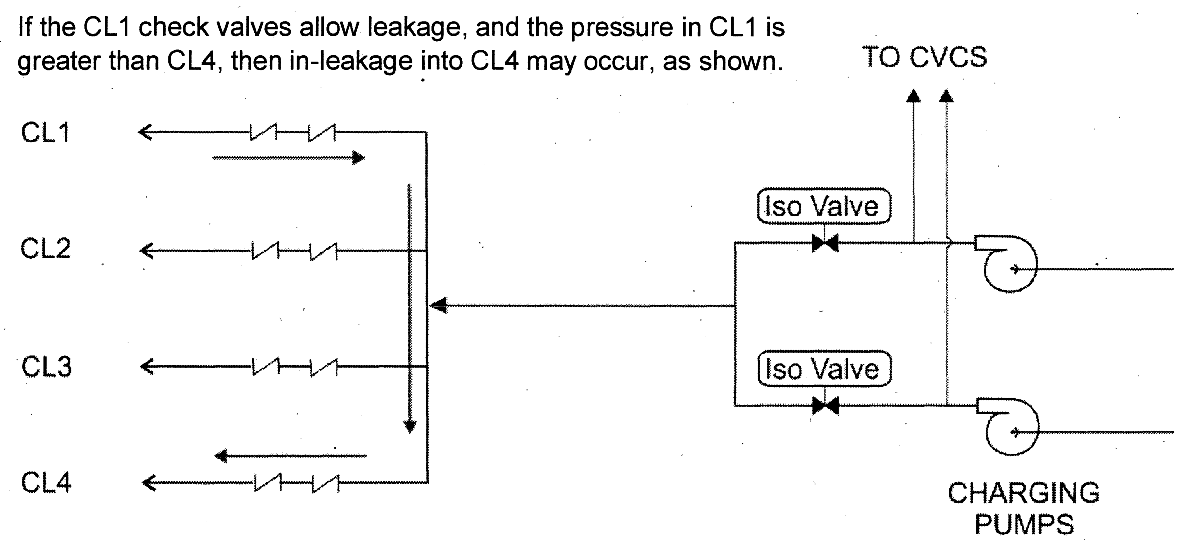
Figure 1: Schematic of a PWR 4-Loop Safety Injection System with Leaking Isolation Valve

Thermal cycling due to in-leakage can result in rapid thermal fatigue damage that can lead to cracking. Dampierre-1 (a French PWR) experienced a through-wall leak in non-isolable safety injection piping attributed to swirl penetration and in-leakage in December 1996. After replacement in-kind of the affected piping, a follow-up examination in October 1997 revealed new thermal fatigue cracking of as much as 33% through-wall depth due to continued in-leakage. Refer to MRP-85 [3] on thermal fatigue OE in RCS piping for additional details.

CROSS-FLOW IN-LEAKAGE CONDITION

In February 2017, leakage was detected in the safety injection piping at McGuire Unit 2, which caused a forced plant shutdown. The cracking occurred in the non-isolable portion of the Loop 4 branch line near the intrados of a 5D bend elbow. The affected line was managed under the plant's MRP-146 program, as it was susceptible to in-leakage. In fact, the line was previously replaced in Spring 2014 due to thermal fatigue cracking at the nozzle weld downstream of the elbow. No indications were found during a follow-on inspection during the Fall 2015 outage.

Subsequent investigation into the cause of the February 2017 through-wall cracking revealed another in-leakage source – cross-flow of reactor coolant into the affected Loop 4 safety injection branch line from one of the adjacent safety injection branch line (Loops 1, 2 or 3). Cross-flow (or cross-leakage) can occur when there is a common header serving multiple trains of a system. If a slightly higher-pressure exists in the RCS loop, and any of the check valve(s) in adjacent loops of the connected piping are leaking, in-leakage may occur into one or more of the other RCS loops (see Figure 2).



Note: Some piping configurations only have single check valve isolation from the RCS.

Figure 2: Schematic of Cross-Flow In-Leakage

The cross-flow phenomenon is not new to the industry, but it has been observed under different circumstances. MRP-85 [3] and the EPRI Fatigue Management Handbook (MRP-235) [4] both discuss cross-flow; however, it has typically only been seen during plant startup or cooldown when less than the full number of reactor coolant pumps are operating. The accompanying rise in static pressure in conjunction with a leaking check valve allows reactor coolant to flow from the loop with the idle RCP, through the check valve, and back toward the

other loops, resulting in potential thermal cycling in both the forward and reverse flow lines (see Figure 3). However, the number of cycles from this scenario is limited since this generally occurs only temporarily during startup or shutdown conditions when a RCP is idle and the available temperature difference is low.

Section 1.2 of MRP-146 [1] describes the same cross-flow condition that was seen at McGuire Unit 2 based on OE from D.C. Cook Unit 1 in 2015 where elevated temperatures were measured in their safety injection header[1]. However, the MRP-146 screening criteria do not address the potential for cross-flow in-leakage and guidance is not provided on how to manage this specific thermal fatigue mechanism.

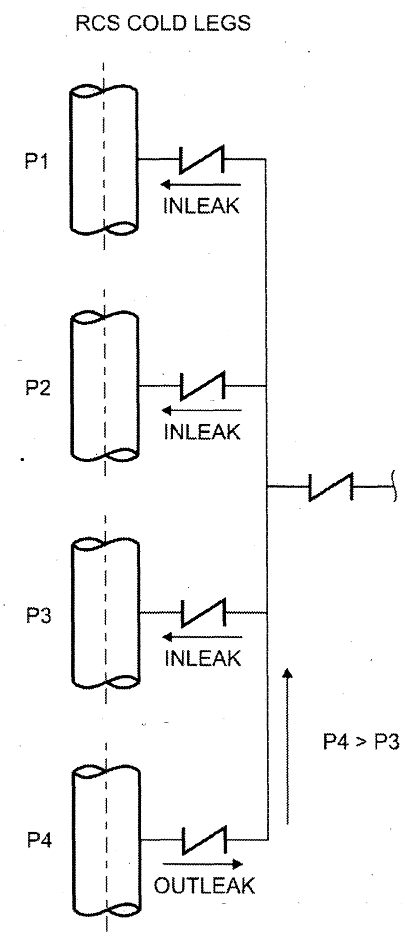


Figure 3: Schematic of Cross-Flow Potential During Startup or Shutdown with Loop 4 RCP Idle

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[1] Note that, during their Fall 2017 refueling outage, D.C. Cook completed their MRP-146 examinations in the affected piping with no findings.

SUMMARY

The purpose of this letter is to inform you of this specific OE that resulted in a forced plant shutdown earlier this year. The MRP TFFG is continuing deliberations on whether any interim guidance is necessary. Based on current industry knowledge, it is likely that the TTFG will provide recommendations for stations to review their safety injection systems for the potential of cross-flow in-leakage and seek opportunities to mitigate risk at their facilities. Plant-specific investigations or augmented examinations may also be recommended.

If you or your team have any follow-up questions, or have any related plant-specific OE to share, please contact Paul Crooker, EPRI TFFG Lead (pcrooker@epri.com; 650-391-7057), Terry Childress, Chairman of the MRP Technical Support Advisory Committee (terry.childress@duke-energy.com, 704-382-5715) or Elliot J. Long, EPRI MRP Technical Support Advisory Committee Lead (elong@epri.com; 412-495-6659).

Sincerely,

Terry Childress
Chair, Technical Support TAC
Duke Energy Corporation

Elliot J. Long
MRP Technical Support TAC Lead
EPRI

cc: MRP TAC Members