Nondestructive Evaluation: License Renewal—Small Bore Piping Evaluation Process

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EPRI Project Manager

P. O'Regan

ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 ▪ PO Box 10412, Palo Alto, California 94303-0813 ▪ USA
800.313.3774 ▪ 650.855.2121 ▪ askepri@epri.com ▪ www.epri.com

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1300 W.T. Harris Boulevard
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Principal Investigator
P. O'Regan

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PRODUCT DESCRIPTION

As part of the application process for license renewal of pressurized water reactors (PWRs) and boiling water reactors (BWRs), U.S. nuclear power plants must perform an evaluation to confirm that they have adequately considered all aging effects that could cause degradation to plant components during the extended period of operation. For all such components, a management plan must be developed and presented to the NRC for approval.

Results and Findings

There are a number of existing industry efforts that address the same issue as the License Renewal (LR) commitment. Indirectly, the industry has already or is in the process of addressing this commitment. Thus, integration and coordination of these activities with the LR commitment will enhance the cost-effectiveness as well as minimize the impact (that is, cost, dose) of meeting this commitment.

Challenges and Objectives

Plant resources are finite. Application of new technology or the extension of existing technology to new plant programs, processes or commitments are challenging. Project objectives include minimizing duplication of effort and streamlining methodologies, thereby reducing the cost of meeting the Class 1 small bore piping inspection commitment.

Applications, Values, and Use

As a result of this project, several possibilities for fulfilling the one time Class 1 small bore piping commitment have been identified. The key to cost-effectively responding to this commitment will be the level of integration with other ongoing pressure boundary integrity efforts (for example, MRP-146, MRP-139, RI-ISI).

EPRI Perspective

EPRI has championed the development and implementation of cost-effective pressure boundary management activities. Actions necessary to fulfill the one time inspection of Class 1 small bore piping is a logical extension of these activities. This report documents how these complementary activities can be coordinated.

Keywords

License renewal
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1 INTRODUCTION

As part of the application process for license renewal of pressurized water reactors (PWRs) and boiling water reactors (BWRs), U.S. nuclear power plants must perform an evaluation to confirm that they have adequately considered all aging effects that could cause degradation to plant components during the extended period of operation. For all such components, a management plan must be developed and presented to the NRC for approval.

The *Generic Aging Lessons Learned (GALL) Report* (NUREG-1801, Rev. 1, Sept. 2005) identifies aging management programs (AMP) for systems, structures, and components that the NRC finds acceptable in meeting license renewal commitments, as required by 10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants.” For small-bore piping, fittings, and branch connections exposed to reactor coolant, Table 1 of the GALL report specifies “One-Time Inspection of ASME Code Class 1 Small-bore Piping (less than NPS 4)” as an acceptable AMP (ID #48 for BWRs and ID #70 for PWRs).

To date, 28 sites (representing 49 units) have either applied or been approved for license renewal. A review of a number of the 17 approved applications indicates that most plants have made this one-time inspection commitment to the NRC.

The objectives of this project were to review the existing small bore commitment, existing industry activities in the LR arena, and identification of relevant industry activities. The project will support cost-effective response to this licensee renewal commitment, minimizing duplication of effort by licensees while streamlining implementation.

This project is a companion project to one that is investigating the NDE capability for small bore piping. See product ID 1015056.

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EVALUATION PROCESS

As previously discussed, most nuclear generating plants that have received U.S. Nuclear Regulatory Commission (NRC) approval for license renewals have committed to one-time inspection of ASME Class 1 small-bore piping (less than NPS 4). This section describes how several plants have implemented programs to address this commitment, status of relevant industry efforts, insights from RI-ISI applications, and how integrating and coordinating these activities can be used to cost-effectively respond to the one time inspection commitment.

Relevant Industry Augmented Inspection Programs

Inservice inspection is typically conducted through implementation of an ASME Section XI program. In addition to these programs, plants have implemented augmented inspection programs in response to regulatory or industry recommended guidance. With respect to the small bore one time inspection commitment, programs addressing the three following degradation mechanisms are of note:

- Primary water stress corrosion cracking (PWSCC, MRP-139)
- Thermal fatigue (PWR:MRP-146, BWR: VIP-155)
- IGSCC in BWRs (VIP-075, Note only applies to piping greater than 4 NPS)

PWSCC has become a major area of effort within the PWR community. A number of studies, inspections and testing have been and are being carried out. In addition, NRC and industry interaction has been frequent and in detail. As such, it appears additional inspections to meet the one time inspection commitment can be met by integration and coordination with this augmented program. It should be noted that the outcome of this material management program may result in actions other than inspections.

As discussed in the operating experience section below, there have been a limited number of thermal fatigue failures in Class 1 small bore piping. In response to these events, the PWR community developed a program to identify susceptible locations and recommend monitoring inspection activities as appropriate [1]. A companion document [2] has also been developed for the BWR community. As such, it appears additional inspections to meet the one time inspection commitment can be met by integration and coordination with this augmented program.

Finally, probably the most intensive augmented inspection program in the US nuclear industry, with the possible exception of flow accelerated corrosion (FAC), has been the BWR fleet's response to Generic Letter 88-01 [NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," January 25, 1988]. While initially developed in the 1980s, these programs have undergone continual updating to reflect lessons learned and operating experience. The most recent guidance is provided in BWRVIP-075 [3]. However, because the scope of the augmented program is only for piping greater than 4 NPS, it is not applicable to the one time inspection commitment for small bore piping. As such, and because there has been some service experience suggesting small bore piping may be susceptible to IGSCC, plants should consider inspections, where applicable.

Relevant Industry Operating Experience

In support of the project, a review of industry operating experience was conducted. This review utilized two types of information sources. The first is a database of piping failures compiled by the OECD Pipe Failure Data Exchange (OPDE) and the other was a review of a number of plant specific service history reviews which were conducted as part of the development of RI-ISI programs.

The OPDE identified that there have been a number of occurrences of failures in small bore Class 1 piping systems. The vast majority of these failures were due to vibratory fatigue, thermal fatigue and stress corrosion cracking (IGSCC, PWSCC, and TGSCC). As seen in the data, as well as generally accepted throughout the industry, vibratory fatigue is the dominant mechanism affecting small bore component reliability.

From an inspection perspective, vibratory fatigue is a mechanism that is not amenable to one time or periodic inspection. The reasoning for this is that vibratory fatigue is a mechanism where the endurance life of the component is essentially used up prior to crack initiation. That is, once the crack has occurred, it propagates quickly to component failure (this is, throughwall). Thus, even if an inspection was conducted for components subjected to this mechanism today, there is no guarantee that failure will not occur in the near future. Thus, other methods (non NDE) are more practical and effective in controlling component reliability for this mechanism [4].

With regards to thermal fatigue, as discussed above there are ongoing industry initiatives that address this mechanism. These efforts are above and beyond that being conducted via the ASME Section XI program or its approved alternative (for example*,* RI-ISI), that almost all plants have implemented.

For the SCC mechanism, IGSCC and PWSCC are dominant. As discussed above, there are industry programs to address IGSCC in BWRs [3, although less than 4 NPS is specifically excluded] and PWSCC [5].

A key component of developing a RI-ISI program involves conducting a plant-specific service history review. The purpose of this review is to collect and evaluate plant-specific operating experience with regards to piping reliability. The outcome of this review is that relevant piping failures have been identified and the piping failure assessment (of the RI-ISI process) has been confirmed as applicable to the particular plant. Additionally, this information is used to support the identification of required inspection locations and applicable degradation mechanisms for which inspections are needed.

As part of this project, a number of these efforts were reviewed for insights as to the one time inspection commitment for small bore piping. This review confirms the findings of that obtained for the OPDE effort discussed above. As an example, a sample of a plant-specific service history review is provided in Appendix A.

Risk-informed Inservice Inspection (RI-ISI)

RI-ISI has been implemented extensively in the US nuclear industry, while there are multiple methodologies, each contain the same fundamental steps as follows:

- Definition of RI-ISI program scope
- Failure Mode and Effects Analysis (FMEA) of pipe segments
- Evaluation of consequences of pipe failures
- Evaluation of pipe failure potential
- Characterization of risk segments
- Inspection element selection
- Evaluation of risk impact of changes to inspection program
- Incorporation of long term RI-ISI program

Approximately ninety percent of the US industry has or is implementing RI-ISI programs. Even if a formal RI-ISI has not been adopted, the generic insights from these evaluation can be used as is discussed in Section 3.

Key considerations in adopting these insights for the one time inspection commitment are:

- Confirmation that the scope of the two programs is consistent. That is, some RI-ISI programs may not encompass all of the piping contained in the one time inspection commitment scope. If not, the generic insights from Section 3 may be used.
- Confirmation that the methodology's failure potential evaluation covers all applicable types of degradation.
- A plant-specific service history review should have been completed.
- Some RI-ISI programs have approval to use visual examination in lieu of volumetric examination for some small bore piping.

LR Small Bore Programs

In support of this project, a review of several plant-specific programs/commitments was conducted. The purpose of this review was to identify the status of plant implementations, develop lessons learned, develop good practice and minimize duplication of effort by the industry.

For each of the plants reviewed, the plant had conducted an assessment of piping failure (that is, consequence of failure) and failure likelihood (for example, potential degradation, CUF). For some plants this was done in a qualitative fashion while other plants had the benefit of a previously conducted RI-ISI program to guide the locations selected for inspections.

Consistent with above, one plant also identified the need to conduct inspections on small bore piping that potentially could be susceptible to IGSCC which is not covered by the Generic Letter 88-01 program.

Based on a review of these programs, the use of risk insights, either qualitative or quantitative, have been beneficial in identifying locations for inspection.

One additional area of interest is the inspection of socket welded connections versus butt welded connections. As seen from the service experience review, vibratory fatigue is the dominant failure mechanism for these types of components although there has been some experience with thermal fatigue [1]. One plant's safety evaluation [6] identifies the following as an acceptable way of fulfilling the one time inspection commitment for small bore piping, including socket welded connections.

> The staff determined that the applicant had committed to do a non-destructive or destructive examination of one socket weld prior to the period of extended operation in response to the staff's concern in this area. As this is a sampling process, the staff determined that one socket weld will represent the population for Class 1 piping less than 4-inch NPS. With this new commitment and the examination of 10 percent of the butt welds in all Class 1 small bore piping, there is reasonable assurance that the aging of small bore piping will be adequately managed during the period of extended operation.

Thus, the NRC has approved an inspection population of ten percent for Class 1 small bore piping butt welds as meeting the one time inspection commitment. In addition, the NRC has approved the use of a single inspection (DE or NDE) as fulfilling the one time inspection commitment for Class 1 small bore socket welds.

3 SUMMARY

Insights from this Review

There are a number of existing industry efforts that address the same issue as the License Renewal (LR) commitment. Indirectly the industry already has or is in the process of addressing this commitment. Thus, integration and coordination of these activities with the LR commitment will enhance the cost-effectiveness as well as minimize the impact (that is, cost, dose) of meeting this commitment.

In particular, most plants have already developed RI-ISI programs for this scope of piping. RI-ISI programs identify the likelihood of piping failures. This is typically done by conducting an assessment as to the type of degradation mechanisms that could be operative in a given piping system. As discussed above, it may be warranted to assure that applicable degradation mechanisms have been evaluated for each application (for example, IGSCC in BWRs). Additionally, even for plants without RI-ISI programs, there are industry activities underway (for example, MRP-139, MRP-146) that have captured industry operating experience with regards to degradation. As such, other than an operating experience review to identify any potential outliers, it appears that this portion of the issue is also being addressed.

From a consequence of failure perspective, as discussed above, most plants have already developed RI-ISI programs for this scope of piping. RI-ISI programs identify the consequence of piping failures. The metrics for this evaluation are the impact on conditional core damage probability (CCDP) and conditional large early release probability (CLERP). For both PWRs and BWRs, postulated piping failures in piping between the RPV and the first isolation valve result in a LOCA while piping between the first and second isolation valve do not. As such, piping between the RPV and the first isolation valve typically have a higher consequence of failure than piping between the two isolation valves. The exception to this is that for some plants (for example, BWRs) there is Class 1 piping that penetrates containment. This piping, if it were to fail, could result in a LOCA outside containment (for example, LERF impact). Although, for this scenario, large bore piping tends to dominate. These types of insights are also generically applicable to plants that have not yet transitioned to a RI-ISI program.

Finally, as mentioned previously, the Oyster Creek Safety Evaluation [6], identified that conducting a ten percent sample for small bore butt welds and examination (NDE or DE) of one socket weld provides an acceptable program to address the one-time inspection commitment for small bore piping. This ten percent population is consistent with many NRC approved RI-ISI applications on Class 1 systems. As such, RI-ISI program results as well as generic risk insights can also be used as an effective means for selecting this one time inspection population.

4 CONCLUSIONS

Based upon the review and investigation conducted by the project, the following conclusions can be made:

- The industry already has in place a number of efforts which can be used to fulfill in part, or in whole, the Class 1 small bore one time inspection commitment (for example, MRP-146, VIP-155, RI-ISI).
- Plants with existing RI-ISI programs can directly use insights from these programs (for example, consequence of failure, potentially operative degradation mechanisms) to identify the one time inspection population.
- Plants without existing RI-ISI programs can use generic risk insights to identify the one time inspection population.
- A ten percent inspection population has been approved by the NRC as an acceptable population size for meeting the one time inspection commitment for Class 1 small bore piping butt welds. It is noted that this sample size (ten percent) is consistent with many previously approved RI-ISI applications on Class 1 piping.
- A single socket weld examination (DE or NDE) has been approved by the NRC as an acceptable population size for meeting the one time inspection commitment for Class 1 small bore socket welds.
- Inspection of some Class 1 small bore BWR locations for IGSCC may be warranted.

5
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2. *BWRVIP-155: BWR Vessel and Internals Project, Evaluation of Thermal Fatigue Susceptibility in BWR Stagnant Branch Lines*. EPRI, Palo Alto, CA: June 2006. 1013389.
3. *BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)*. EPRI, Palo Alto, CA: October 1999. TR-113932.
4. *EPRI Fatigue Management Handbook, Volumes 1 through 3*. EPRI, Palo Alto, CA. TR-104534.
5. *Materials Reliability Program: Primary System Piping Butt Weld Inspection and Evaluation Guideline (MRP-139)*. EPRI, Palo Alto, CA. 1010087.
6. USNRC Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station, March 2007.

A

EXAMPLE PLANT SPECIFIC SERVICE HISTORY REVIEW

Table A-1
Service History and Susceptibility Review Results for the Reactor Coolant System

Source Documents / Databases Reviewed for	Damage Mechanisms													O.D.
Historical Piping Pressure Boundary	**Thermal Fatigue**		**Stress Corrosion Cracking**				**Localized Corrosion**			**Flow Sensitive**		**Mechanical**	**Other**	**Initiated**
Degradation Occurrences at ANO-1	**TASCS**	**TT**	**IGSCC**	**TGSCC**	**ECSCC**	**PWSCC**	**MIC**	**PIT**	**CC**	**E-C**	**FAC**	**VF**	**Findings**	
Station Information Management System	None	None	None	None	None	None	None	None	None	None	None	**PBF(3)**	**(1)** **PD(2)**	**No**
Paperless Condition Reporting System	**PE(4)**	None	None	None	None	None	None	None	None	None	None	**PBF(3)**	None	**No**
Licensing Research System	**PE(4)**	None	None	None	None	**PBF(5)**	None	None	None	None	None	**PBF(3)**	**PD(2)**	**No**
Nuclear Plant Reliability Database System	None	None	None	None	None	None	None	None	None	None	None	None	None	**No**
ANO-1 ISI Program Records	None	None	None	None	None	None	None	None	None	None	None	None	**(1)**	**No**
Control Room Station Log	None	None	None	None	None	None	None	None	None	None	None	None	None	**No**
System Upper Level Documents	None	None	None	None	None	None	None	None	None	None	None	None	None	**No**
Other Station Documents	**P(6)**	**P(6)**	None	None	None	**P(6)**	None	None	None	None	None	None	None	**No**

Legend:

P (Precursor) - This category includes identification of postulated damage mechanisms and loadings through knowledge of operating parameters, water chemistry, and so on. No physical evidence of pressure boundary degradation currently exists. This category includes postulated mechanisms identified as a result of this review.

PE (Plant Event) - This category includes identification of postulated damage mechanisms and loadings as a result of an observed or potential plant event (for example, water hammer). No physical evidence of pressure boundary degradation currently exists.

PD (Physical Damage) - This category includes identification of observed pressure boundary degradation as evidenced by cracking, pitting, wastage, thinning, physical deformation or other deterioration.

PBF (Pressure Boundary Failure) - This category includes identification of through-wall flaws resulting from the effects of an identified damage mechanism.

Notes:

1. Reference JO 00770489 and ISI program records. Multiple indications (surface and subsurface) have been identified over time in the reactor coolant system. These indications were either removed (for example, gouges or linear surface flaws) or evaluated (for example, laminar or planar subsurface flaws) and determined to be Code acceptable. None of these indications were attributed to an inservice damage mechanism and are believed to have been non-service induced (that is, fabrication or other origin).

2. Reference JO 00723183, LER 86-006 and IE Information Notice 86-108, which document corrosion wastage (boric acid) on the exterior of the P-32A discharge cold leg HPI nozzle region due to leakage from an above HPI isolation valve (body-to-bonnet leak).
3. Reference JO 00776670, JO 00776959, JO 00786058, CR 1-89-0029, CR 1-89-0312, CR 1-90-0010, LER 89-002, LER 89-010 and ANO correspondence to the NRC 1CAN107414, 1CAN107420, 1CAN027502 and 1CAN107507 which document small diameter (1½ and 1 inch NPS) cold leg drain line leaks primarily attributable to vibrational fatigue.
4. Reference NRC Bulletin 88-08, NRC Bulletin 88-11, CR C-88-0047, CR 1-92-0327, CR 1-93-0164, CR 1-98-0117 and ANO correspondence to the NRC 0CAN019102, 0CAN088912, 0CAN108806, 0CAN109104, 0CAN119007, 1CAN038903, 1CAN039101, 1CAN068906, 1CAN079201, 1CAN108914, 1CAN128910 and 1CAN129105 which address the potential for thermal stratification in the reactor coolant system.
5. Reference LER 90-021, which documents a small diameter (1 inch NPS) pressurizer level tap nozzle leak attributed to primary water stress corrosion cracking.
6. Reference Calculation No. EPRI-116-310 of the ANO-1 RI-ISI pilot application submittal, which identifies the potential for TASCS, TT and PWSCC in the reactor coolant system.

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3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 • USA
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