# Nondestructive Evaluation: RI-ISI Status, Extensions, and Resolution of Implementation Issues—2009

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# PRODUCT DESCRIPTION

This report describes the results of the risk-informed in-service inspection (RI-ISI) Support project for the first three quarters of 2009. The RI-ISI Support project provides for the coordination and participation of risk technology to various industry groups and identifies and extends the use of risk technology to other components and programs.

## Results and Findings

This report describes the results of a project developed to provide industry RI-ISI support. This project provides for the coordination and participation of risk technology to various industry groups: ASME, Materials Reliability Program (MRP) (for example, Fatigue and Alloy 600/182/82 Task Groups), Boiling Water Reactor Vessel Internals Project (BWRVIP), Equipment Reliability, Nuclear Energy Institute (NEI) RI-ISI Working Group, Advisory Committee on Reactor Safeguards )ACRS) , regulatory bodies, owners groups, International Atomic Energy Agency (IAEA), European Network on Inspection Qualification (ENIQ), and Nuclear Regulators Working Group (NRWG) interactions. In addition, this project identifies and extends the use of risk technology to other components and programs. This project also addresses minimizing the impact of emerging issues on plant staff—incorporating industry experience into the RI-ISI methodologies—and interaction with industry and regulatory bodies.

## Challenges and Objectives

Because plant resources are finite, the application of new technology or the extension of existing technology to new plant programs and processes is challenging. Objectives of this project include minimizing duplication of effort, streamlining risk-informed methodologies in order to reduce the cost of implementation and maintenance, and identifying future potential applications of the technology.

## Applications, Value, and Use

Lessons learned from pilot plant applications are being incorporated into generic methodologies for use by the industry as a whole. In addition, extensions to the existing technology are being developed to address other components and plant programs.

## EPRI Perspective

The Electric Power Research Institute (EPRI) has championed the use of risk-informed technology to increase the reliability of pressure boundary integrity, reduce low value requirements, and streamline the implementation of this technology. This report documents the continuation of that effort.

## Approach

Extension in using risk-informed technologies requires that methodologies and tools be developed that can be cost-effectively applied and maintained. In addition, appropriate stakeholders (for example, plant operators, regulators, standards development organization, and technical support organizations) must be engaged so that short- and long-term benefits of this technology can be realized.

**Keywords**
Risk-informed
In-service inspection
Risk-informed in-service inspection (RI-ISI)
Probabilistic risk assessment (PRA)
Probabilistic safety assessment (PSA)
Repair and replacement

# ACKNOWLEDGMENTS

The Electric Power Research Institute (EPRI) would like to acknowledge the support and guidance provided by the NDE Program Integration Committee, Action Plan Committee. Their continued support provides the foundation for the development and application of this technology to the vast majority of U.S. plants as well as several international members. In addition, Arturo Smith from Entergy and Sam Volk from Progress Energy were key sponsors for this effort and provided invaluable insight and direction.

# CONTENTS

# 1

# INTRODUCTION

This report describes the results of a project developed to provide industry risk-informed in-service inspection (RI-ISI) support. This project provides for the coordination and participation of risk technology to the following industry groups:

- ASME
- Materials Reliability Program (for example, Fatigue and Alloy 600/182/82 Task Groups)
- Boiling Water Reactor Vessel Internals Project (BWRVIP)
- Equipment Reliability
- Nuclear Energy Institute RI-ISI Working Group (NEI)
- Advisory Committee on Reactor Safeguards (ACRS)
- Regulatory bodies
- Owners groups
- International Atomic Energy Agency (IAEA)
- European Network on Inspection Qualification (ENIQ)
- Nuclear Regulators Working Group (NRWG)

This project also identifies and extends the use of risk technology to other components and programs.

This project also addresses minimizing the impact of emerging issues on plant staff, incorporating industry experience into the RI-ISI methodologies, and interacting with industry and regulatory bodies. This report describes the progress of the first three quarters of 2009.

# 2
# ASME CODE ACTIVITIES

This project supports several activities being processed through the ASME code, including Section XI and Section III as well as the Board of Nuclear Codes and Standards Task Group on Risk Management and the Working Group on New LWRs. Many of these activities involve pilot plant activities or are in formative stages.

## Streamlined RI-ISI (Code Case N716)

In the vein of continuous improvement, the Electric Power Research Institute (EPRI) Nondestructive Evaluation (NDE) Program has developed a streamlined RI-ISI methodology: risk-informed/safety-based ISI (RIS_B). The benefits of this new approach are that the risk-informed technology can be applied and maintained more cost-effectively and to the whole plant (that is, a full-scope application) while at the same time providing significant cost, worker exposure, and radwaste savings.

To demonstrate the methodology, two pilot plant applications were conducted and submitted to the U.S. Nuclear Regulatory Commission (NRC; the Commission) for its review and approval. The first application was to a BWR (Grand Gulf), and the second was to a two-unit PWR (Cook Nuclear Plant). The Grand Gulf application was approved on September 21, 2007, and the DC Cook approval occurred on September 28, 2007. A third application conducted at Waterford (PWR) was approved in April 2008. These applications are termed *new plant applications* in that they are transitioning from a deterministic ISI program to a risk-informed ISI program.

Since the approval of the three applications just described, several plants with previously approved RI-ISI programs have decided to transition to EPRI's streamlined approach. The rationale for this transition is the ease of implementation and maintenance as well as the reduction in probabilistic risk assessment (PRA) resources needed to support the streamlined approach. EPRI staff have participated in these transition applications and are using the insights from these efforts to incorporate lessons learned into a revision of the Code Case. In addition, this revision to the Code Case extends the scope of application beyond piping welds to other pressure boundary component inspections (for example, examination categories C-A, C-B, C-D, and C-G). Not only will this revision increase the benefit of this technology (for example, the elimination of undue burden), but it will provide for a more consistent ISI program. That is, currently plants may have an ISI program defined for piping welds based on risk-informed technology while adjacent components are still founded on their original, deterministic program. This revision to the Code Case has been approved by the Working Group on the Implementation of Risk-Based Examinations (WGIRBE) and Subgroup Water Cooled Systems (SGWCS). A copy of the current draft of this revision is contained in Appendix A.

### Risk-Informed Repair and Replacement

In support of extending risk-informed technology to repair and replacement activities as well as support for other related industry initiatives, Arkansas Nuclear One, Unit 2 (ANO-2) volunteered to serve as an industry pilot plant for the EPRI-developed methodology. This methodology has been reflected in various draft Codes Cases, was submitted to the NRC, and was recently approved in April 2009. The methodology can now be used by ANO-2 staff to conduct risk-informed repair and replacement activities. Use of this methodology will significantly reduce the cost of repair and replacement activities on low-safety-significant components because Section XI requirements will no longer apply.

Although the safety-related, low-safety-significant items (also known as *RISC-3 components*) are removed from the scope of Section XI requirements, the methodology continues to impose some high-level treatment controls onto these components. These high-level treatment controls are intended to mirror existing balance-of-plant controls, as defined by the licensee, and will provide “reasonable confidence” that these components will continue to satisfy their design functional requirements when demanded. The controls are intended to include relevant testing, inspections, and corrective action programs to the extent necessary to show that assumptions in the classification process remain valid.

EPRI has work under way to take the pilot plant results and lessons learned and develop a process that can be used for industry-wide implementation. This work has resulted in Code Case N752, which has been balloted and approved by the Working Group on the Implementation of Risk-Based Examinations (WGIRBE) and Working Group Design and Programs (WGDP).

# 3

# IMPACT OF REGULATORY GUIDE 1.200, REVISION 1

As presented in Regulatory Information Summary 2007-06, over the past 30 years the NRC and the nuclear industry have performed many PRAs for various purposes—with the scope, depth, and technical content dictated by the specific purpose and use. To encourage the use of PRA in all regulatory matters, the NRC in 1995 issued a policy statement on the use of PRA methods in NRC activities. Since then, the NRC has used PRAs for many applications, including generic safety issue prioritization, regulatory analysis in support of rulemaking and backfits, reactor oversight and inspection programs, and risk-informed regulation. At the same time, the nuclear industry increasingly uses PRA to support its operating and licensing processes, including risk-informed license amendment requests, relief requests, configuration risk management, and design certification and licensing of new reactors. As a result, PRAs are becoming a mainstream regulatory tool and provide valuable input into the decision-making process for plant design, operation, and maintenance. Consequently, confidence in the information derived from a PRA must be high. The scope of the analysis must be sufficiently broad and the accuracy of the technical content of sufficient rigor to justify the specific results and insights from the PRA supporting the decision under consideration.

On December 18, 2003, the Commission issued a directive, “Stabilizing the PRA Quality Expectations and Requirements.” In this Staff Requirements Memorandum (SRM), the Commission approved the implementation of a phased approach to achieving an appropriate quality for PRAs for NRC risk-informed regulatory decision making. The Commission described the phased approach in an attachment to the SRM and directed the staff to develop an action plan that would define a practical strategy for its implementation.

To this end, Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 1” (RG 1.200, Revision 1) has been issued by the NRC. Since January 2008, the NRC has been using this Regulatory Guide for reviewing risk-informed applications, including RI-ISI applications. Although this Regulatory Guide is not applicable to existing approved RI-ISI relief requests, it is applicable for all new relief requests as well as those plants requesting re-approval (that is, for the next 10-year inspection interval).

Industry and NRC have met specifically on RI-ISI and discussed the extent to which RG 1.200 is required for RI-ISI applications. Currently, the plant-specific PRA that supports the RI-ISI application must either meet the Regulatory Guide or provide justification for alternatives.

In response to this need, the 2008 EPRI report *Nondestructive Evaluation: PRA Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs* (1018427) was submitted to the NRC. This report provides guidance on PRA technical adequacy needed to support a RI-ISI application. After the NRC has approved this report, a licensee can review its PRA against the guidelines in EPRI report 1018427, confirm that its PRA meets this guidance, and use it as justification in a RI-ISI submittal—thereby eliminating requests for additional information (RAIs) related to PRA technical adequacy. As of the writing of this report, the NRC had issued RAIs on EPRI report 1018427, which are presented in Appendix B. Responses to these RAIs are under development and will be submitted to the NRC in the fourth quarter of 2009.

# 4

# INTERNATIONAL ACTIVITIES

In addition to support provided to the U.S. nuclear fleet, several activities have been under way in the international community. These activities include support of the Risk-Informed In-Service Methodologies (RISMET) project and the ENIQ's Task Group on Risk (TGR) as well as the IAEA.

## Risk-Informed In-Service Methodologies

The Organisation for Economic Cooperation and Development's Committee on the Safety of Nuclear Installations and the EU's Joint Research Center have embarked on a RI-ISI benchmark study. The overall objective of this study is to apply various RI-ISI methodologies to the same case (that is, selected piping systems in one nuclear power plant). This comparative study aims to identify the main differences in the methodologies and their impact on the final results. Regulators, plant operators, and other interested stakeholders from several countries—Belgium, Canada, the Czech Republic, France, Germany, Japan, Sweden, Switzerland, Germany, and the United States—are participating.

EPRI participation includes review and critique of applications of various RI-ISI methodologies to a single unit. In particular, EPRI participation will ensure that the EPRI methodology is used and understood correctly. EPRI staff are also applying the EPRI RI-ISI methodology to the host plant.

Applications of various methodologies have been made, and a review of results is under way—as are interactions between the "application groups" and the "evaluation groups." A final report is under development.

## European Network on Inspection Qualification Task Group on Risk

ENIQ is a European organization, under which the TGR issued the report *European Framework Document for Risk-Informed In-Service Inspection* (ENIQ Report nr. 23, EUR 21581 EN). This high-level document is intended to provide guidelines to utilities for developing their RI-ISI approaches and using or adapting established approaches to the European environment, taking into account utility-specific characteristics and national regulatory requirements.

Although the focus of the TGR has been on the RISMET project previously described, current tasks being worked on or evaluated by this group are the following:

- Interaction between RI-ISI and qualification or probability of detection curves
- Benchmarking RI-ISI methodologies
- Sensitivity and uncertainties analyses (SRMs/probabilistic safety assessment [PSA])
- Criteria for risk importance measures and risk acceptance criteria
- Justification of partial scope RI-ISI application

- Code Case N716: applicability in EU
- Guidelines for expert panels
- Guidelines for use of PSA in RI-ISI
- Defense in depth
- Expert elicitation for degradation mechanisms
- Interaction between SRM databases and SRM code verification and validation
- RI-ISI application for internals of reactor pressure vessels
- Feedback on inspection outcome
- Design-based classification
- Risk monitors

## International Atomic Energy Agency

As part of the IAEA mission to support its member countries, a technical document on RI-ISI is being developed and is scheduled to be published in late 2009. EPRI NDE Program staff served as lead authors of this report. The subject Table of Contents for the report is provided in Appendix C, and a brief summary of the report and its intended purpose is provided next.

> During the design phase of the first nuclear power plants, it was believed that the high standards used to design and fabricate passive components would allow a problem-free operation throughout their lifetime. For this reason, the need for in-service inspection was not considered. However, when plants became operational, it was discovered that components still degraded over time despite such high design standards, and the industry began to develop inspection programs. ASME developed a standard, the ASME Boiler and Pressure Vessel Code, Section XI: Rules for Inservice Inspection of Nuclear Power Plant Components, which initially provided rules for inspection of Class 1 systems only. Over 30 years the Code was revised to address many needs including the inspection of Class 2 and 3 systems.
>
> Traditionally, a number of commercial nuclear power plants implemented ASME Section XI to ensure the structural integrity of systems. Section XI was based on a sampling approach: 25% of Class 1 and 7.5% of Class 2 piping welds were examined to verify that no generic degradation existed. To search for generic degradation, Section XI required that piping be examined based on materials, configuration, and potential stress levels. These criteria, although useful as inputs for determining possible examination locations, were not suited to be used alone as selection criteria. Because of these inadequacies, problems were typically identified via non-Section XI activities, for example, operator walkdowns or augmented programs. Consequently, nuclear plants were devoting significant staff time, radiation exposure, and financial resources to examine locations with low failure potential and/or little safety significance.
>
> In the past, while Section XI Inservice Inspection was based on a sampling approach, WWER operating countries followed Russian rules, standards and experience of designers and equipment manufacturers. The Russian approach was based in general on hydrostatic pressure tests due to the lack of reliable NDT volumetric methods. A too-high reliance on the experience of manufactures led to partially non-systematic development

of ISI programs as different manufacturers and designer groups applied their specific philosophies, taking into account preferably their own practical manufacturing rules and experience not integrated into an overall ISI program development approach. ISI programs that were developed in such a way became mandatory ISI programs approved by the Regulatory Bodies of individual countries. For this reason, there were no accepted sampling rules for piping systems, and a variation in the percentage and locations of piping welds selected for examination could occur. In recent years, this situation has become more specific in different countries. On one side, there are ISI programs that strictly follow Russian standards; on the other side, there are ISI programs following new national standards and approaches that take into account worldwide experience and standards.

Traditional ISI requirements looked for generic degradation. Industry experience has shown that degradation is typically not a random occurrence. Degradation occurs where the conditions necessary for a particular mechanism exist. Over time a better understanding of the degradation typically found in the systems at a nuclear power plant has been developed. A potential degradation mechanism can be assigned to those locations where the appropriate conditions may exist. Thus, locations that have a higher failure potential can be targeted.

Risk-informed technology allows plants to take the next step and inspect those systems or portions of systems that are most risk important. Risk-informed in-service inspection (RI-ISI) reflects recent developments in probabilistic safety assessment (PSA) technology, structural reliability, and the experience gained from nearly 13,000 reactor years operating experience of nuclear power plants. *Risk* is defined in the engineering community as the product of the consequences of a failure and the probability of that failure occurring. Using RI-ISI, the risk significance of a component and its failure potential are determined. This allows the plant to target its resources to examine locations that are truly risk significant, providing the ability to capture or minimize risk and thereby improving plant reliability while keeping radiation doses to workers as low as reasonably achievable.

This document introduces the general approach of RI-ISI technology, application status in member states, discussion of related issues, and ongoing research developments including applications for new plants. This document provides guidance on RI-ISI technology and informs readers of the current status and ongoing R&D activities. In Section 2, a generic approach is given to RI-ISI. Section 3 introduces the overview of various RI-ISI methodologies and the current application status. Section 4 describes the process and organization. Section 5 highlights a number of topical issues that should be addressed during any attempt to apply RI-ISI. The document concludes with a summary of activities that are being undertaken in the world (Section 6).

# 5
# SUMMARY

This report describes the results of the project titled “RI-ISI Support” for the first three quarters of 2009. The RI-ISI Support project provides for the coordination and participation of risk technology to various industry groups and identifies and extends the use of risk technology to other components and programs.

# *A*

# DRAFT OF REVISION 1 TO CODE CASE N716

## Code Case N-716-1 Alternative Piping System Classification and Examination Requirements, Section XI Division 1

**Applicability**: 1989 Edition through the 2007 Edition with the 2009 Addenda

**Inquiry:** What alternative to the requirements to IWB-2420, IWB-2430 and IWB-2500 (Examination Categories B-F and B-J), and IWC-2420, IWC-2430 and IWC-2500 (excluding Examination Categories C-C and C-H), or as additional requirements for Subsection IWD, may be used for inservice inspection and preservice inspection of Class 1, 2, 3, or Non-Class pressure retaining components?

**Reply:** It is the opinion of the Committee that the following requirements may be used in lieu of the requirements of IWB-1220, IWB-2420, IWB-2430, Table IWB-2500-1 (Examination Categories B-F and B-J), IWC-1220, IWC-2420, IWC-2430, Table IWC-2500-1 (excluding Examination Categories C-C and C-H), for inservice inspection of Class 1 or 2 pressure retaining components , IWB-2200, and IWC-2200 for preservice inspection of Class 1 or 2 pressure retaining components, or as additional requirements for Class 3 pressure retaining components or Non-Class pressure retaining components for plants issued an initial operating license prior to December 31, 2008, plant designs in operation prior to December 31, 2008 or approved via the USNRC design certification process (10CFR50 Part 52).

### 1.0 SCOPE

The scope shall include Class 1 and 2 pressure retaining components, as identified in IWB-1200 and IWC-1200, Components Subject to Examination limited by the exemptions of IWB-1220 and IWC-1220. Depending upon the results of 2(a)(5) below, the provisions of this Case may define additional requirements for Class 1 or 2 pressure retaining components exempt from NDE by Section XI, Class 3 or Non-Class pressure retaining components.

### 2.0 GENERAL REQUIREMENTS

*(a)* Pressure retaining components shall be assigned a category that shall be used to determine the treatment requirements of this Case. High safety significant pressure retaining components consist of pressure retaining components that are

*(1)* Class 1 portions of the reactor coolant pressure boundary (RCPB), except as provided in (c)(2)(i) and (c)(2)(ii) of Title 10 of the U.S. Code of Federal Regulations (10 CFR), Part 50.55a,

*(2)* applicable portions of the normal shutdown cooling pressure boundary function shall be included. That is, Class 1 and 2 pressure retaining components of systems or portions of systems needed to utilize the normal shutdown cooling flowpath either

*(a)* as part of the RCPB from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds, or

*(b)* other systems or portions of systems from the reactor pressure vessel (RPV) to the second isolation valve (i.e., farthest from the RPV) capable of remote closure or to the containment penetration, whichever encompasses the larger number of welds,

(*3)* that portion of the Class 2 feedwater system [> NPS 4 (DN 100)] of pressurized water reactors (PWRs) from the steam generator, including the steam generator, to the outer containment isolation valve,

(4) piping within the break exclusion region [> NPS 4 (DN 100)] for high energy piping systems(1) as defined by the Owner, and

(5) any piping segment or component whose contribution to core damage frequency is greater than 1E-06, or whose contribution to large early release frequency is greater than 1E-07, based upon a plant-specific probabilistic risk assessment (PRA) of pressure boundary failures (e.g., pipe whip, jet impingement, spray, and inventory losses). This may include exempt , Class 3, or Non-Class pressure retaining components.

*(b)* Low safety significant pressure retaining components shall include all other Class 1, 2, 3, or Non-Class pressure retaining components not classified as high safety significant in accordance with this Case.

## 3.0 PRESERVICE EXAMINATION REQUIREMENTS

As an alternative to the preservice examination requirements of IWB-2200, IWC 2200, or IWD-2200, as applicable, the following requirements apply to HSS pressure retaining components.

(a) Initial Examination

(1) Pressure retaining components, other than piping welds, classified as HSS require preservice inspection. The examination volumes, areas, techniques, and procedures shall be in accordance with the applicable requirements of Section XI.

(2) Twenty five percent of the piping weld population classified as high safety significant require volumetric pre-service inspection. This population includes those HSS piping welds selected for volumetric inservice inspection by this case. If piping welds requiring volumetric inservice inspection do not equal twenty five

percent, additional piping welds shall be selected for pre-service inspection. These additional piping welds shall be selected using the criteria of Section 4 of this Case, to the extent practical. The examination volumes, areas, techniques, and procedures shall be in accordance with the applicable requirements of Table 1 of this Case.

(3) No preservice examination for LSS pressure retaining components is required.

(b) Repair / Replacement Activities

(1) Prior to return to service following a repair/replacement activity on HSS pressure retaining components, other than for piping welds, examinations shall be performed in accordance with the applicable requirements of Section XI. Prior to return to service following a repair/replacement activity on HSS piping welds, examinations shall be performed to meet the requirements of 3(a)(2).

(2) For pressure retaining components that become HSS following a reevaluation as a result of a repair/replacement activity, the required first inservice examination performed in accordance with this Case shall serve as the preservice examination.

(3) No preservice examination for LSS pressure retaining components is required.

## 4.0 INSERVICE INSPECTION REQUIREMENTS

Low safety significant pressure retaining components are exempt from the volumetric, surface, VT-1 and VT-3 visual examination requirements of Section XI.

High safety significant pressure retaining components, other than piping welds, shall be selected in accordance with Section XI for examination. The examination requirements for these locations are defined in Section XI.

Ten percent of the high safety significant piping welds shall be selected for examination. The examination requirements for these locations are defined in Table 1.

The existing plant FAC inspection program and localized corrosion inspection program, excluding crevice corrosion (per Table 2), shall not be credited toward the ten percent requirement.

The existing plant IGSCC (Category B through G) inspection program may be credited toward the 10 % requirement, provided the requirements of this Case are met.

Selection of piping welds for examination shall be as follows:

*(a)* The susceptibility of each high safety significant item to the degradation mechanisms listed in Table 2 shall be determined. High safety significant piping welds shall be assigned an item number in Table 1 based upon the results of the degradation mechanism evaluation. High safety significant piping welds identified as not susceptible shall be assigned to Item No. R1.20 of Table 1.

*(b)* Examinations shall be prorated equally among systems to the extent practical, and each system shall individually meet the following requirements:

*(1)* A minimum of 25 % of the population identified as susceptible to each item number and item number combination (e.g., R1.11 and R11.16) shall be selected, excluding item numbers R1.18 and R1.20.

*(c)* For the RCPB;

*(1)* ten percent of the piping welds shall be selected.

*(2)* at least two thirds of the examinations shall be located between the first isolation valve (i.e., isolation valve closest to the RPV) and the reactor pressure vessel. If this can not be accomplished due to a limited number of available welds between the first isolation valve and the RPV, then 25% of the piping welds between the RPV and the first isolation valve shall be selected, not to exceed ten of ten percent of the RCPB.

*(d)* A minimum of 10% of the piping welds in that portion of the RCPB that lies outside containment (e.g., portions of the main feedwater system in BWRs) shall be selected.

*(e)* For each system within the break exclusion region 10 % of the piping welds shall be selected.

*(f)* If the examinations selected above exceed ten percent of the total number of high safety significant piping welds, the examinations may be reduced by prorating among the requirements of (b), (c), (d) and (e) above, to the extent practical, such that at least ten percent of the high safety significant population is inspected.

*(g)* If the examinations selected above are not at least ten percent of the high safety significant piping weld population, additional welds shall be selected so that the total number selected for examination is at least ten percent of the high safety significant population. The additional piping welds may be selected using the criteria of (b), (c), (d) and (e) above.

*(h)* When selecting welds for examination, the following shall be considered:

- *(1)* plant-specific cracking experience,
- *(2)* weld repairs,
- *(3)* random selection,
- *(4)* minimization of worker exposure.

*(i)* For raw water systems(2) identified as susceptible to localized corrosion, excluding crevice corrosion, in lieu of the ten percent inspection population above, an inspection population and successive examination requirements shall be determined per section 3.6.7.1 or section 3.6.7.2 of EPRI TR-112657, Revision B-A(3).(j) For plant designs approved via the USNRC design certification process (10CFRPart52) and containing material that has not had significant operating time (i.e. > ten years) in a nuclear plant environment applicable for its intended application (e.g. alternatives to A82/182 in a reactor coolant environment versus chrom-moly in a feedwater/steam cycle environment), the inspection population size shall be 25% of the locations (e.g. welds) containing the material.

## 5.0 CHANGE-IN-RISK EVALUATION

A change-in-risk evaluation shall be performed prior to the initial implementation of this Case.

*(a)* *Bounding Failure Frequency.* The failure frequencies of 2E-06 per weld-year for welds in the high failure potential category, 2E-07 per weld-year for welds in the medium failure potential category, and 1E-08 per weld-year in the low failure potential category may be used as bounding failure frequencies for piping welds as defined in Table 3. Failure frequencies for non piping welds may be obtained from industry sources such as EPRI TR-102266 and ERPI TR-1012302. Additionally, probabilistic fracture mechanics (PFM) tools may be used. If these PFM tools are used, the requirements of Appendix II shall be met.

*(b)* *Conditional Risk Estimates.* The estimated conditional core damage probability (CCDP) and conditional large early release probability (CLERP) may be used if available. Bounding values of the highest estimated CCDP and CLERP may be used if specific estimates are not available.

*(c)* The following general equations shall be used to estimate the change-in-risk. One estimate shall be made for the change in core damage frequency (CDF) and one for large early release frequency (LERF). The equations only illustrate the change in CDF. The change in LERF due to application of the process shall be estimated by substituting the CLERP for CCDP in the equations.

$$\Delta R_{CDF} = \Sigma_j (I_{rj} - I_{ej})*PF_j*CCDP_j$$

where,

$\Sigma_j$ = summation of locations selected for examination

$\Delta R_{CDF}$ = change in CDF due to replacing the prior deterministic ISI program with the ISI program developed in accordance with this Case

$I_{rj}$ = factor of reduction in pipe rupture frequency at location "j" associated with the ISI program developed by this Case

$I_{ej}$ = factor of reduction in pipe rupture frequency at location "j" associated with the prior deterministic ISI program

$PF_j$ = piping failure frequency at location "j" without examination

$CCDP_j$ = conditional core damage probability at location "j"

In terms of probability of detection
( $POD_j = (1 - I_j)$ ), the equation becomes

$$\Delta R_{CDF} = \sum_j (POD_{ej} - POD_{rj})*PF_j*CCDP_j$$

where,
$POD_{ej}$ = probability of detection at location "j" associated with the prior deterministic ISI program

$POD_{rj}$ = probability of detection at location "j" associated with the ISI program developed in accordance with this Case

It is acceptable to use bounding estimates for pipe failure frequency, conditional core damage probability, and conditional large early release probability to simplify the calculations. If the bounding estimates for pipe failure frequency and conditional probability are used, the equation becomes:

$$\Delta R_{CDF} = [(POD_e*N_{efc} - POD_r*N_{rfc})]*PF_f* CCDP_c$$

where,
$POD_e$ = probability of detection in the existing ISI program (may be degradation mechanism specific)

$N_{efc}$ = number of examination locations in the consequence "f" and failure frequency "c" categories associated with the prior deterministic ISI program [need not include locations with a surface only examination and identified as not susceptible to outside diameter attack per the criteria of Table 2]

$POD_r$ = probability of detection in the ISI program developed by this Case (may be degradation mechanism specific)

$N_{rfc}$ = number of examination locations in the consequence "f" and failure frequency "c" categories associated with the ISI program developed by this Case

$PF_f$ = piping failure frequency for the high, medium, and low failure frequency estimates

$CCDP_c$ = conditional core damage probability consequence estimates

*(d) Acceptance Criteria.* Any increase in CDF and LERF for each system shall be less than 1E-07 per year and 1E-08 per year respectively, and the total increase in CDF and LERF should be less than 1E-06 per year and 1E-07 per year respectively. If necessary, additional examinations shall be selected to meet this acceptance criteria.

## 6.0 SUCCESSIVE INSPECTIONS AND ADDITIONAL EXAMINATIONS

Examination Category R-A Welds shall meet the successive inspection and additional examination requirements of (a) and (b) below. All other examination categories shall meet the applicable successive inspection and additional examination requirements of Section XI.

*(a)* SUCCESSIVE INSPECTIONS

As an alternative to the successive inspection requirements of IWB-2420, IWC-2420, or IWD-2420, the following requirements shall be met. Successive examinations for item number R1.18 are outside the scope of this Case.

*(1)* The sequence of piping examinations established during the first inspection interval using this Case shall be repeated during each successive inspection interval to the extent practical. The examination sequence may be modified to optimize scaffolding, radiological, insulation removal, or other considerations, provided the percentage requirements of Table IWB-2411-1 or IWB- 2412-1 are met.

*(2)* If piping structural elements are accepted for continued service by analytical evaluation in accordance with IWB-3132.4 or IWB-3142.4 before, during, or after implementation of this Case, the areas containing flaws or relevant conditions shall be reexamined during the next three inspection periods.

*(3)* If the reexaminations required by 6*(a)(2)* reveal that the flaws or relevant conditions remain essentially unchanged for three successive inspection periods, the examination schedule shall revert to the original schedule of successive inspections.

*(b)* ADDITIONAL EXAMINATIONS

As an alternative to the additional examination requirements of IWB-2430, IWC-2430, or IWD-2430, the following requirements shall be met. Additional examinations for item number R1.18 are outside the scope of this Case.

*(1)* Examinations performed in accordance with Table 1 of this Case, excluding item number R1.18, that reveal flaws or relevant conditions exceeding the acceptance standards of Table IWB- 3410-1 shall be extended to include a first sample of additional examinations during the current outage.

*(a)* The piping structural elements (welds) to be examined in the first sample of additional examinations shall include HSS elements with the same postulated degradation mechanism in systems whose materials and service conditions are similar to the element that exceeded the acceptance standards.

*(b)* The number of examinations required is the number of HSS elements with the same postulated degradation mechanism scheduled for the current inspection period. If there are not enough HSS elements to equal this number, the Owner shall include remaining HSS elements and LSS elements up to and including this number that are subject to the same degradation mechanism.

*(2)* If the additional examinations required by 6*(b)(1)* reveal flaws or relevant conditions exceeding the acceptance standards of Table IWB- 3410-1, the examinations shall be extended to include a second sample of additional examinations during the current outage.

*(a)* The second sample of additional piping structural elements to be examined shall include all remaining HSS piping structural elements in Table 1 subject to the same degradation mechanism.

*(b)* The Owner shall also examine LSS piping structural elements subject to the same degradation mechanism or document the basis for their exclusion.

*(3)* For the inspection period following the period in which the examinations of 6*(b)(1)* and *(2)* were completed, the examinations shall be performed as originally scheduled in accordance with IWB-2400.

## 7.0 PROGRAM UPDATES

Examination selections made in accordance with this Case shall be reevaluated on the basis of inspection periods that coincide with the inspection program requirements for Inspection Program A or B of IWA-2431 or IWA-2432 (1989 Edition through 2004 Edition with 2006 Addenda), as applicable. For Inspection Program B, the third period reevaluation will serve as the subsequent inspection interval reevaluation. As of the 2007 Edition of ASME Section XI, Inspection Program A was eliminated and Inspection Program B was addressed under IWA-2431. As such, for the 2007 Edition through the latest Edition and Addenda, examination selections made in accordance with this Case shall be reevaluated on the basis of inspection periods that coincide with the inspection program requirements for IWA-2431. For the inspection program, the third period reevaluation will serve as the subsequent inspection interval reevaluation.

*(a)* plant design changes (e.g., physical: new piping or equipment installation; programmatic: power uprating / 18 to 24 month fuel cycle; and procedural: operating procedure changes)

*(b)* changes in postulated conditions or assumptions (e.g., check valve seat leakage greater than previously assumed)

*(c)* examination results (e.g., discovery of leakage or flaws)

*(d)* Failures in HSS or LSS pressure retaining components (e.g., plant-specific or industry occurrences of through-wall or through-weld leakage, failure due to a new degradation mechanism, or a non-postulated mechanism)(4)

*(e)* PRA updates that would increase the scope of (2)*(a)(5)* (e.g., new initiating events, new system functions, more detailed model used, and initiating event and failure data changes)

*(f)* the impact of 7*(a)* through 7*(e)* on the change-in-risk evaluation in 5.

## 8.0 OWNER'S RESPONSIBILITY

*(a)* The Owner shall determine of the appropriate classification for pressure retaining components in accordance with the provisions of this Case.

*(b)*

The Owner shall ensure that the technical adequacy of the PRA shall be reviewed to confirm it meets the requirements of Appendix I of this Case or the owner has met PRA technical adequacy requirements for RI-ISI applications accepted by the regulatory body having jurisdiction at the plant site.

*(c)* The results of the application of this Case (e.g., determination of high safety significant pressure retaining components , change in risk evaluation) shall be documented and reviewed.

Notes:

(1) NUREG-0800, section 3.6.2 provides a method for defining this scope of piping.

(2) The term Raw Water System is a general term to define a secondary or tertiary cooling water system that does not contain nuclear grade water (e.g. primary or secondary). Plant specific identifiers for these types of systems include normal service water, emergency service water, nuclear service water, RHR service water and circulating water. These systems may or may not be subject to water treatment (e.g. use of chlorine, biocides).

(3) EPRI TR-112657, Revision B-A, Revised Risk-informed Inservice Inspection Evaluation Procedure, December, 1999.

(4) NUREG-0737 (Clarification of TMI Action Plan Requirements) provides an acceptable method for review of industry operating experience.

# TABLE 1
# EXAMINATION CATEGORIES

| EXAMINATION CATEGORY R-A | | | | | | | |
|---|---|---|---|---|---|---|---|
| | | | | | Extent and Frequency [Note (3)] | | |
| Item No. | Parts Examined | Examination Requirement/ Fig. No. [Note (2)] | Examination Method | Acceptance Standard | 1st Interval | Successive Intervals | Defer to End of Interval |
| R1.10 | High Safety Significant Piping Structural Elements | | | | | | |
| R1.11 | Elements Subject to Thermal Fatigue | IWB-2500-8(c) [Note (1)]<br>IWB-2500-9, 10, 11 | Volumetric [Note (8), (10)] | IWB-3514 | Element [Notes (2), (4)] | Same as 1st | Not Permissible |
| R1.12 | Not Used | | | | | | |
| R1.13 | Elements Subject to Erosion Cavitation | [Note (6)] | Volumetric [Note (7)] | IWB-3514 [Note (6)] | Element [Note (2)] | Same as 1st | Not Permissible |
| R1.14 | Elements Subject to Crevice Corrosion Cracking | [Note (5)] | Volumetric [Notes (9), (10)] | IWB-3514 | Element [Note (2)] | Same as 1st | Not Permissible |
| R1.15 | Elements Subject to Primary Water Stress Corrosion Cracking (PWSCC) | IWB-2500-8(c) [Note (1)]<br>IWB-2500-9, 10, 11 | Volumetric [Notes (7), (9), (10)] | IWB-3514 | Element [Notes (2), (4)] | Same as 1st | Not Permissible |
| R1.16 | Elements Subject to Intergranular or Transgranular Stress Corrosion Cracking (IGSCC or TGSCC) | IWB-2500-8(c) [Note (1)]<br>IWB-2500-9, 10, 11 | Volumetric [Notes (7), (9), (10)] | IWB-3514 | Element [Notes (2), (4)] | Same as 1st | Not Permissible |
| R1.17 | Elements Subject to localized corrosion [Microbiologically Influenced Corrosion (MIC) or Pitting] | IWB-2500-8(a),<br>IWB-2500-8(b),<br>IWB-2500-8(c),<br>IWB-2500-9, 10, 11 | Visual, VT-3 Internal Surfaces or Volumetric [Notes (6) or (7)] | [Note (6)] | Element [Note (2)] | Same as 1st | Not Permissible |
| R1.18 | Elements Subject to Flow Accelerated Corrosion (FAC) | [Note (7)] | [Note (7)] | [Note (7)] | [Note (7)] | [Note (7)] | [Note (7)] |
| R1.19 | Elements Subject to External Chloride Stress Corrosion Cracking (ECSCC) | IWB-2500-8(a),<br>IWB-2500-8(b),<br>IWB-2500-8(c),<br>IWB-2500-9, 10, 11 | Surface | IWB-3514 | Element [Note (2)] | Same as 1st | Not Permissible |
| R1.20 | Elements not Subject to a Degradation Mechanism | IWB-2500-8(c)<br>IWB-2500-9, 10, 11 | Volumetric [Notes (9), (10)] | IWB-3514 | Element [Notes (2), (4)] | Same as 1st | Not Permissible |

NOTES:

(1) The length of the examination volume shown in Figure IWB-2500-8(c) shall be increased by enough distance [approximately ½ in. (13mm)] to include each side of the base metal thickness transition or counterbore transition.

(2) Includes examination locations and Class 1 weld examination requirement figures that typically apply to Class 1, 2, 3, or Non-Class welds identified in accordance with paragraph (4) Inservice Inspection Requirements.

(3) Includes essentially 100% of the examination location. When the required examination volume or area cannot be examined due to interference by another component or part geometry, limited examinations shall be evaluated for acceptability. Acceptance of limited examinations or volumes shall not invalidate the results of the change-in-risk evaluation (see 5). Areas with acceptable limited examinations, and their bases, shall be documented.

(4) The examination shall include any longitudinal welds at the location selected for examination in Note (2). The longitudinal weld examination requirements shall be met for both transverse and parallel flaws within the examination volume defined in Note (2) for the intersecting circumferential welds.

(5) The examination volume shall include the volume surrounding the weld, weld HAZ, and base metal, where applicable, in the crevice region. Examination should focus on detection of cracks initiating and propagating from the inner surface.

(6) The examination volume shall include base metal, welds, and weld HAZ in the affected regions of carbon and low alloy steel, and the welds and weld HAZ of austenitic steel. Examinations shall verify the minimum wall thickness required. Acceptance criteria for localized thinning is in course of preparation. The examination method and examination region shall be sufficient to characterize the extent of the element degradation.

(7) In accordance with the Owner's existing programs such as PWSCC, IGSCC, MIC, or FAC programs, as applicable.

(8) Socket welds of any size and branch pipe connection welds NPS 2 (DN 50) and smaller selected for examination require a volumetric examination of the piping base metal within ½ inch (13 mm) of the toe of the weld and the fitting itself shall receive a VT-2 visual examination.

(9) Socket welds of any size and branch pipe connection welds NPS 2 (DN 50) and smaller require only a VT-2 visual examination. For PWSCC susceptible locations, the insulation shall be removed.

(10) The VT-2 visual examination is to be conducted during a system pressure test or a pressure test specific to that element or segment, in accordance with Examination Categories B-P, C-H or D-B as applicable.

## TABLE 2
## DEGRADATION MECHANISMS

| **Mechanisms** (1) | | **Attributes** | **Susceptible Regions** |
|---|---|---|---|
| *TF* | *TASCS* | – piping > NPS 1 (DN 25); and<br>– pipe segment has a slope < 45 deg from horizontal (includes elbow or tee into a vertical pipe), and<br>– potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids, or<br>potential exists for leakage flow past a valve (i.e., in-leakage, out-leakage, cross-leakage) allowing mixing of hot and cold fluids, or<br>potential exists for convection heating in dead-ended pipe sections connected to a source of hot fluid, or<br>potential exists for two phase (steam / water) flow, or<br>potential exists for turbulent penetration in branch pipe connected to header piping containing hot fluid with high turbulent flow, and<br>– calculated or measured $\Delta T > 50$°F (28°C), and<br>– Richardson number > 4.0 | nozzles, branch pipe connections, safe ends, welds, heat affected zones (HAZ), base metal, and regions of stress concentration |
| | *TT* | – operating temperature > 270°F (130°C) for stainless steel, or<br>operating temperature > 220°F (105°C) for carbon steel, and<br>– potential for relatively rapid temperature changes including<br>cold fluid injection into hot pipe segment, or<br>hot fluid injection into cold pipe segment, and<br>– $\lvert \Delta T \rvert > 200$°F (110°C) for stainless steel, or<br>$\lvert \Delta T \rvert > 150$°F (83°C) for carbon steel, or<br>$\lvert \Delta T \rvert > \Delta T$ allowable (applicable to stainless and carbon) | |
| *SCC* | *IGSCC (BWR)* | – evaluated in accordance with existing plant IGSCC program per NRC Generic Letter 88-01, or alternative (e.g., BWRVIP-075) | austenitic stainless steel welds and HAZ |
| | *IGSCC (PWR)* | – operating temperature > 200°F (93°C), and<br>– susceptible material (carbon content $\geq$ 0.035%), and<br>– tensile stress (including residual stress) is present, and<br>– oxygen or oxidizing species are present<br>OR<br>– operating temperature < 200°F (93°C) , the attributes above apply, and<br>– initiating contaminants (e.g., thiosulfate, fluoride, chloride) are also required to be present | |
| | *TGSCC* | – operating temperature > 150°F (65°C), and<br>– tensile stress (including residual stress) is present, and<br>– halides (e.g., fluoride, chloride) are present, or<br>caustic (NaOH) is present, and<br>– oxygen or oxidizing species are present (only required to be present in conjunction w/halides, not required w/caustic) | austenitic stainless steel base metal, welds, and HAZ |

## TABLE 2 (Cont'd)

### *DEGRADATION MECHANISMS*

| **Mechanisms** | | **Attributes** | **Susceptible Regions** |
|---|---|---|---|
| *SCC* | *ECSCC* | – operating temperature $>$ 150°F (65°C), and<br>– an outside piping surface is within five diameters of a probable leak path (e.g., valve stems) and is covered with non-metallic insulation that is not in compliance with Reg. Guide 1.36, or an outside piping surface is exposed to wetting from concentrated chloride-bearing environments (e.g., seawater, brackish water, brine) | austenitic stainless steel base metal, welds, and HAZ |
| | *PWSCC* | – piping or weld material is UNS N06600, N06082, or W86182, and<br>– exposed to primary water at T $>$ 570°F (300°C), and<br>– the material is mill-annealed and cold worked, or cold worked and welded without stress relief | nozzles, welds, and HAZ without stress relief |
| *LC* | *MIC* | – operating temperature $<$ 150°F (65°C), and<br>– low or intermittent flow, and<br>– pH $<$ 10, and<br>– presence/intrusion of organic material (e.g., raw water system), or water source is not treated w/biocides (e.g., refueling water tank) | fittings, welds, HAZ, base metal, dissimilar metal joints (e.g., welds, flanges), and regions containing crevices |
| | *PIT* | – potential exists for low flow, and<br>– oxygen or oxidizing species are present, and<br>– initiating contaminants (e.g., fluoride, chloride) are present | |
| | *CC* | – crevice condition exists (i.e. thermal sleeves), and<br>– operating temperature $>$ 150°F (65°C), and<br>– oxygen or oxidizing species are present | |
| *FS* | *E-C* | – existence of cavitation source (i.e., throttling or pressure reducing valves or orifices)<br>– operating temperature $<$ 250°F (120°C), and<br>– flow present $>$ 100 hrs/yr, and<br>– velocity $>$ 30 ft/s (9.1 m/s), and<br>– $(P_d - P_v) / \Delta P < 5$ where, $P_d$ = static pressure downstream of the cavitation source, $P_v$ = vapor pressure, and $\Delta P$ = pressure difference across the cavitation source | fittings, welds, HAZ, and base metal |
| | *FAC* | – evaluated in accordance with existing plant FAC program | per plant FAC program |

LEGEND:

Thermal Fatigue (TF)
Thermal Stratification, Cycling, and Striping (TASCS)
Thermal Transients (TT)
Stress Corrosion Cracking (SCC)
Intergranular Stress Corrosion Cracking (IGSCC)
Transgranular Stress Corrosion Cracking (TGSCC)
External Chloride Stress Corrosion Cracking (ECSCC)
Primary Water Stress Corrosion Cracking (PWSCC)
Localized Corrosion (LC)
Microbiologically Influenced Corrosion (MIC)
Pitting (PIT)
Crevice Corrosion (CC)
Flow Sensitive (FS)
Erosion-Cavitation (E-C)
Flow Accelerated Corrosion (FAC)

## TABLE 3
## DEGRADATION MECHANISM CATEGORY

| **Failure Potential** | **Conditions** | **Degradation Category** | **Degradation Mechanism** |
|---|---|---|---|
| High [NOTE 1] | Degradation mechanism likely to cause a large break | Large Break | Flow-Accelerated Corrosion |
| Medium | Degradation mechanism likely to cause a small leak | Small Leak | Thermal Fatigue, Erosion-Cavitation, Corrosion, Stress Corrosion Cracking |
| Low | None | None | None |

*Note:*
*(1) Segments having degradation mechanisms listed in the small leak category shall be upgraded to the high failure potential/large break category if the pipe segments also have the potential for water hammer loads.*

# APPENDIX I

# PRA Technical Adequacy Requirements

| **Supporting Requirement as defined in USNRC Regulatory Guide 1.200, r1 (1)** | **Capability Category Required** |
|---|---|
| | |
| IF-D5A. | **III & II** |
| | |
| IE-A3a, IE-C11, IE-C12, AS-A7, SY-A7, SY-A15, SY-B2, HR-D7, HR-F1, DA-B2, DA-C9, , IF-D3a, IF-E3a. | **II & I** |
| | |
| IE-A4, IF-C3, IF-C6, IF-C8 | **II** |
| | |
| IE-E4a, IE-A6, IE-A7, IE-B3, AS-A9, AS-A10, SC-A2, SC-A5, SC-B1, SC-B2, SY-A4, SY-A20, SY-B1, SY-B7, SY-B11, HR-B1, HR-C2, HR-D2, HR-D3, HR-E3, HR-E4, HR-F2, HR-G1, HR-G3, HR-G4, HR-G5, HR-H1, DA-B1, DA-C7, DA-C8, DA-C10, DA-C12, DA-D1, DA-D3, DA-D4, DA-D5, DA-D6, DA-D7, IF-A1a, IF-C3b, IF-D3, QU-A2b, QU-D3, QU-D5a, QU-E3, QU-F3, LE-B1, LE-B2, LE-C1, LE-C2a, LE-C2b, LE-C3, LE-C4, LE-C8a, LE-C8b, LE-C9a, LE-C9b, LE-C10, LE-D1a, LE-D1b, LE-D2, LE-D3, LE-D4, LE-D5, LE-D6, LE-E2, LE-E3, LE-F1a, LE-G3. | **I** |
| | |
| IE-A10, IE-B5, IE-C1, IE-C1a, IE-C2, IE-C3, IE-C4, IE-C5, IE-C13, IE-D3, AS-C3, SC-C3, SY-C3, HR-I3, DA-E3, IF-D6, QU-E1, QU-E2, QU-E4, QU-F4, LE-F2, LE-F3, LE-G4. | **Need not be met** |
| | |
| IE-A1, IE-A2, IE-A3, IE-A5, IE-B1, IE-B2, IE-B4, IE-C1b, IE-C6, IE-C7, IE-C8, IE-C9, IE-C10, IE-D1, IE-D2, AS-A1, AS-A2, AS-A3, AS-A4, AS-A5, AS-A6, AS-A8, AS-A11, AS-B1, AS-B2, AS-B3, AS-B4, AS-B5, AS-B5a, AS-B6, AS-C1, AS-C2, SC-A1, SC-A4, SC-A4a, SC-A6, SC-B3, SC-B4, SC-B5, SC-C1, SC-C2, SY-A1, SY-A2, SY-A3, SY-A5, SY-A6, SY-A8, SY-A10, SY-A11, SY-A12, SY-A12a, SY-A12b, SY-A13, SY-A14, SY-A16, SY-A17, SY-A18, SY-A18a, SY-A19, SY-A21, SY-A22, SY-B3, SY-B4, SY-B5, SY-B6, SY-B8, SY-B10, SY-B12, SY-B13, SY-B14, SY-B15, SY-B16, SY-C1, SY-C2, HR-A1, HR-A2, HR-A3, HR-B2, HR-C1, HR-C3, HR-D1, HR-D4, HR-D5, HR-D6, HR-E1, HR-E2, HR-G2, HR-G6, HR-G7, HR-G9, HR-H2, HR-H3, HR-I1, HR-I2, DA-A1, DA-A1a, DA-A2, DA-A3, DA-C1, DA-C2, DA-C3, DA-C4, DA-C5, DA-C6, DA-C11, DA-C11a, DA-C13, DA-C14, DA-C15, DA-D2, DA-D6a, DA-D8, DA-E1, DA-E2, IF-A1, IF-A1b, IF-A3, IF-A4, IF-B1, IF-B1a, IF-B1b, IF-B2, IF-B3, IF-B3a, IF-C1, IF-C2, IF-C2a, IF-C2b, IF-C2c, IF-C3A, IF-C3c, IF-C4, IF-C4a, IF-C5, IF-C5a, IF-C7, IF-C9, IF-D1, IF-D4, IF-D5, IF-D7, IF-E1, IF-E3, IF-E4, IF-E5, IF-E5a, IF-E6, IF-E6a, IF-E6b, IF-E7, IF-E8, IF-F1, IF-F2, IF-F3, QU-A1, QU-A2a, QU-A3, QU-A4, QU-B1, QU-B2, QU-B3, QU-B4, QU-B5, QU-B6, QU-B7a, QU-B7b, QU-B8, QU-B9, QU-C1, QU-C2, QU-C3, QU-D1a, QU-D1B, QU-D1C, QU-D4, QU-D5b, QU-F1, QU-F2, QU-F5, QU-F6, LE-A1, LE-A2, LE-A3, LE-A4, LE-A5, LE-B3, LE-C5, LE-C6, LE-C7, LE-E1, LE-E4, LE-F1b, LE-G1, LE-G2, LE-G5, LE-G6. | **Spans all three categories and needs to met** |

**Notes:**

1. For purposes of RI-ISI, capability category relate to technical aspects of the plant PRA and as such, peer review findings and / or gaps related to documentation that do not impact the results would allow the capability category to still be considered met.

# APPENDIX II

# PFM User Training and Qualification

To ensure that the input parameters are consistently assigned and the Probabilistic Fracture Mechanics (PFM) model used in this Case is properly executed, the users of the PFM tool shall be trained and qualified. Acceptable qualification and the scope of training is to be determined by the Owner based on the background and experience of the individuals using the tool. Training should consist of the following topics:

*(a)* Overall risk-informed ISI process;

*(b)* How PFM calculated probabilities are used in this Case;

*(c)* Capabilities and limitations of PFM model

*(d)* Expertise and type of information required, including applicable sources;

*(e)* How potential degradation mechanisms are considered and combined;

*(f)* The importance of each input parameter on each degradation mechanism and failure mode;

*(g)* Examples of PFM tool use for different degradation mechanisms and failure modes; and

*(h)* How detailed PFM input (e.g., uncertainties) is developed and used.

# *B*

# RAIS RELATED TO EPRI REPORT 1018427

NRC DRAFT

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# REQUEST FOR ADDITIONAL INFORMATION

# NONDESTRUCTIVE EVALUATION: PROBABILISTIC RISK ASSESSMENT TECHNICAL ADEQUACY GUIDELINES FOR RISK-INFORMED IN-SERVICE INSPECTION PROGRAMS

The staff has reviewed the EPRI Report, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidelines For Risk-Informed In-Service Inspection Programs," 1018427 (Topical), and finds that additional information is needed before we can complete the review. The Topical references the Probabilistic Risk Assessment (PRA) Standard (ASME RA-sb-2005) that was prepared by ASME in 2005 as endorsed by Regulatory Guide 1.200 Revision 1 in 2007, with respect to PRA technical adequacy.

1. The Topical fails to provide general guidelines which describe the overarching framework from which acceptable capability categories (CCs) for individual supporting requirement (SRs) for the internal events PRA can be determined. An example of a general guideline that is included is the Topical's explanation that SRs that solely address quantitative attributes are of limited importance. The risk ranking and change is risk estimates in EPRI's risk-informed inservice inspection (RI-ISI) methods use an order of magnitude approach which reduces the influence of PRA elements that might only change the quantitative results slightly. However, other general elements such as importance of logic modeling and human actions in the internal events PRA should be likewise generally characterized. For example, it would appear that the internal event PRA logic models need to be of relatively high quality (i.e., accurate and high resolution) because multiple consequential SSCs failures need to be evaluated using these logic models. Please identify general guidelines for the technical elements that compose an internal events Level 1/LERF PRA based on how EPRI's RI-ISI method relies on these elements.

2. Due to the lack of general guidelines, many of the discussion on individual SR's appear to be simply conclusions with no justification. Based on the general guidelines developed for RAI 1, please revaluate target categories for the specific SR's in the internal events PRA and indentify which general guideline supports the selected category.

3. The Topical only provides guidance in defining the applicable ASME PRA Standard supporting requirements (SRs) and the appropriate capability category (CC) for the Levels 1 and 2 analyses of internal events while at power. The EPRI report states that, "As future revisions to RG 1.200 occur, this work will be updated to support future RI-ISI application and maintenance."

   a) It is acknowledged that ASME and ANS have issued a combined standard "ASME/ANS RA-Sa-2009" in February 2009 and endorsed in RG 1.200 Revision 2 in March 2009. EPRI should provide its position on this combined standard in support of the RI-ISI PRA technical adequacy including the following hazard groups:

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- Internal Fires
- Seismic Events
- High Winds
- External Floods, and
- Other External Hazards

b) Discuss whether the guidance provided in the Topical would be treated differently for operating plants and plants licensed under Part 52.

4. Page V, second paragraph "Results and Findings," the second sentence states that "The technical adequacy of the PRA is determined by demonstrating that the PRA meets technical elements and associated supporting requirements (SRs) of NRC RG 1.200." It should be noted that RG 1.200 provides the NRC position on supporting requirements (including clarifications as needed), but does not provide supporting requirements as stated in the above statement. Clarify the above sentence.

5. Page V, last paragraph "EPRI Perspective" states that "The vast majority of U.S. plants that implement RI-ISI programs have used tools and products developed by the EPRI. This report reviews these tools and products against the ASME PRA Standard and the NRC RG 1.200." Define "tools and products" mentioned in this paragraph.

6. Based on the review of previous RI-ISI submittals that are based, in part, on ASME Code Case N-716, the NRC staff believes that additional work may be needed beyond the CC recommended in the Topical in order too provide confidence that all high-safety-significant (HSS) segments will be indentified and that an appropriate change in risk is estimated.

   The Topical proposes CC I/II as being sufficient for SR IF-D3a (IFEV-A3). Capability Category I/II permits grouping or subsuming flood initiated scenarios with existing plant initiating event (IE) groups. Capability category III does not permit subsuming flood IEs with other initiators. A RI-ISI analysis may be done long after the flooding analysis is completed and subsuming flood scenarios into existing plant IE groups would require an extra step in the RI-ISI analyses to retrieve any subsumed scenarios. This requirement is mentioned in the table in Appendix A in the Topical. Please propose changes to the N-716 methodology which clarifies this additional step or change the recommended CC to Category III.

   The Topical propose CC II as being sufficient for SR IF-C6 (IFSN-A14) and IF-C8 (IFSN-A16). Capability category II permits screening-out of flood areas and sources respectively based on, in part, the success of human actions to isolate and terminate the flood before equipment is damaged. Capability Category III does not permit screening out areas and sources based on reliance on operator actions. Qualitative screening of flood scenarios based on possible human intervention does not appear to be fully consistent with the CCDP/CLERP or

CDF/LERF significant determination. The Topical simply states that the qualitative screening provides confidence in the high reliability of the human actions. Please explain how the qualitative screening in CC II provides confidence that the quantitative guidelines will not be exceeded or change the recommended CC to Category III.

The Topical propose CC I as being sufficient for SR IF-D5a (IFEV-A6) when determining IE flooding frequencies. Capability category I permits determining flooding IE frequencies using solely generic experience. Capability category II/III requires using a combination of generic and plants specific experience. RI-ISI is directed toward inspecting locations with the highest risk, driven mostly by failure frequency. Code Case N-716 requires even greater fidelity in the IE frequency because it is identifies HSS segments based on risk. The Topical states that CC I is sufficient because that level of detail is sufficient to identify the relative importance of the segments. The staff disagrees. RI-ISI is developed on a plant specific basis using the absolute importance of segments to risk. Pleases further explain why plant operating experience reflected in the flooding failure frequencies is not judged to be important in the RI-ISI process or change the recommended category to II/III.

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