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Living Program Guidance To Maintain Risk-Informed Inservice Inspection Programs For Nuclear Plant Piping Systems

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This document was prepared by the NEI risk-informed inservice inspection task force, including representatives from reactor operating companies, ASME committees, EPRI, and Westinghouse.

NOTICE

EXECUTIVE SUMMARY

The Code of Federal Regulations references American Society of Mechanical Engineers (ASME) Code Requirements, or alternatives endorsed by the Nuclear Regulatory Commission, as a means to address periodic inspections of piping systems and components. Risk-informed methods have been developed and approved by the NRC to allow alternatives to the deterministic inspection requirements of Section XI of the ASME Code. These methods have been implemented at most U.S. nuclear power plants. NRC safety evaluations approving plant implementation have generally included discussion of the need to evaluate the program periodically with regard to its input assumptions and inspection result history. This document provides guidance with regard to considerations for this process.

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LIVING PROGRAM GUIDANCE TO MAINTAIN RISK-INFORMED INSERVICE INSPECTION PROGRAMS FOR NUCLEAR PLANT PIPING SYSTEMS

1 INTRODUCTION

Most U.S. nuclear power plants have implemented risk-informed inservice inspection (RI-ISI) programs. This document provides guidance for considerations in maintaining these programs over time and discusses factors that might influence the program, such as plant modifications, inspection results or changes to the plant probabilistic risk analysis (PRA).

2 BACKGROUND

This section provides background information for developing and maintaining deterministic Section XI (SXI) Inservice Inspection (ISI) programs. Background information also is provided on the development, trial application, NRC approval and subsequent industry implementation of RI-ISI programs. Finally, the report includes pertinent information regarding industry programs for monitoring pressure boundary integrity regardless of whether a plant has implemented a RI-ISI program.

2.1 DETERMINISTIC SXI PROCESS

ISI programs are developed based upon a ten-year inspection interval. Each ten-year ISI program is submitted to the NRC. Exceptions to the endorsed version of the ASME code referenced in 10CFR50 require an NRC-approved relief request.

These programs define ISI requirements, including those for selecting and examining components, non-destructive examination (NDE), pressure testing and repair / replacement activities. As stated above, these programs are updated every 10 years to incorporate newer versions of the ASME code.

2.2 PERSPECTIVE ON RI-ISI PROGRAMS

RI-ISI programs provide alternative selection criteria for the number, location and NDE technique and examination volume for piping components. In addition, many RI-ISI applications are partial scope (e.g. Class 1 only), thus only the risk-informed portion of the overall ISI program would be impacted by the RI-ISI evaluations, while the remainder of the piping would continue to meet deterministic SXI requirements.

As part of implementing a RI-ISI program, plants have committed to maintaining a living RI-ISI program, including a commitment to review the risk ranking at least once per

inspection period. In its simplest term, risk ranking consists of the combination of failure potential and consequence of failure. Thus, a confirmation that pressure boundary integrity continues to be monitored and that the assumed consequence(s) of component failure is unchanged would meet the intent of this commitment.

It is important to note that typically the living program requirement of risk-informed applications, above and beyond existing plant practices, is a function of the risk associated with the particular application. Plants have implemented a variety of risk applications, some of which apply to all plants, and others which are voluntary. Mandatory applications include the individual plant examination (IPE and IPE for external events), maintenance rule (monitoring and configuration risk assessment), and the process for determining the significance of regulatory findings. Voluntary applications include RI-ISI, risk-informed inservice testing, risk-informed technical specification allowed outage times, other risk-informed technical specification initiatives (missed surveillances, mode restraints), Appendix J option B, risk-informed deferrals of integrated leak rate testing, and deletion of hydrogen recombiners.

With the wide variety of applications as noted above, there is a commensurate spectrum of risk impacts. Some applications (e.g. maintenance rule configuration risk assessment) have more significant impacts on plant risk, while others have relatively smaller impacts. Within this spectrum, risk-informed ISI is considered to have a relatively small impact on overall plant risk [core damage frequency (CDF) and large early release frequency (LERF)]. This small risk impact has been acknowledged by the NRC staff and the NRC's Advisory Committee on Reactor Safeguards, and it is one reason that risk-informed ISI was among the first risk applications approved for industry adoption.

Most risk applications involve determination of delta risk, commensurate with the guidance of NRC Regulatory Guide 1.174. This determination is a snapshot of the delta risk for that particular change to the licensing basis at the time of initial approval. Other applications, such as technical specifications, do not involve requirements that the delta risk impact be recalculated and demonstrated to be maintained over the life of the plant. Other controls, such as the maintenance rule, help maintain the basic assumptions that drive the PRA results and the delta risk impacts of applications over time. Further, delta risk impacts of any voluntary application are generally constrained to the "very small change" region of the Regulatory Guide 1.174 risk metric guidelines, which is acknowledged to be conservative with respect to public health impacts as defined in the NRC Safety Goal Policy Statement (Reference 8). This small risk impact was an important factor in determining the appropriate update guidance of this document.

Although it is important to provide guidance on the control of RI-ISI programs over time, and this is in fact required by most NRC safety evaluation reports granting approval for specific plants, it should be noted that the overall risk impact is low. Flexibility should be

provided for reasonable consideration of program impacts over time, based on factors that might affect the original approval basis.

3 CHANGES THAT CAN IMPACT RI-ISI PROGRAMS

This section assesses the potential impact of plant changes on RI-ISI programs. First, a review and assessment was conducted of a number of inputs used in developing RI-ISI programs. Second, the overall RI-ISI process was evaluated with the goal of identifying those plant changes that could impact the RI-ISI evaluations and determining what impact they would have on plant safety. Third, a review of the technical elements comprising a PRA was conducted to contrast their impact on a living RI-ISI program and to identify what impact these changes would have on overall plant safety.

3.1 RI-ISI UNIQUE INPUTS

Table 3-1 contains a listing of a number of references and inputs that could be used in typical RI-ISI applications. The inputs listed in this table are those inputs that are used to support the RI-ISI evaluations and typically would not be used in developing a deterministic Section XI program. Reviewing these types of inputs allows one to assess the RI-ISI program from the bottom up and helps in understanding how they are used in the RI-ISI evaluation process. The review provides insight into the importance from a plant safety perspective of developing additional plant controls, as well as a sense of the benefit of any additional burden imposed by an update requirement.

One insight is that shutdown risk management controls are important. These controls are imposed through the regulatory requirements of 10 CFR 50.65(a)(4), as addressed by NRC Regulatory Guide 1.182, which, in turn endorses industry guidance originally developed through NUMARC 91-06 (Reference 9) for assuring control of risk during shutdown evolutions

3.2 DISPOSITION TABLE

Table 3-2 contains a review of those plant activities that could impact a RI-ISI program. This review is more of a top-down approach, as compared to Section 3.1. This type of review allows one to investigate changes to plant conditions, configurations, procedures and processes and determine if they would impact a RI-ISI program and, if so, what is the existing control process that assures that plant safety is maintained.

As can be seen from this table, many aspects of the RI-ISI program and its basis are currently addressed by existing plant controls and processes. An example exception to this conclusion has to do with surveillance test intervals (STIs) for standby pumps. Many RI-ISI applications credit the existing STIs as pressurizing connected piping, thereby identifying whether leakage is occurring. These test intervals can be weekly, monthly or

quarterly. Obviously, more frequent testing reduces the exposure time for these components. However, the converse is also true. That is, less frequent testing may increase the exposure time for certain components.

Thus, it is recommended that licensees review changes to the parameters listed in Table 3-2 to assess their impact on the RI-ISI program.

3.3 PRA TECHNICAL ELEMENTS

Both sections 3.1 and 3.2 look at the RI-ISI program from a physical/implementation perspective. That is, what plant changes (e.g. hardware, procedures) can cause a change in the number or type of inspections? This section (3.3) assesses the impact of an evolving PRA, revised to reflect updated plant information or PRA methodology, changes in software or modeling changes on the RI-ISI program.

Table 3.3 presents the results of this review by PRA technical element. These technical elements are similar to those contained in PRA Peer Review Process (NEI 00-02; Reference 10) and the ASME PRA standard (RA-S-2002; Reference 11).

3.4 PLANT-SPECIFIC UPDATE EXPERIENCE

A number of plants have completed RI-ISI program updates. This section describes several of these updates, the resultant change to the RI-ISI programs and the basis for the changes identified. These updates cover a spectrum of plants types, scope of application and RI-ISI methodologies.

The detailed results of these updates are provided as Appendix A to this report, and a summary level of the results is provided in this section.

PLANT A

Plant A is an approved Class 1-only application that utilized the Electric Power Research Institute (EPRI) methodology. After the RI-ISI evaluation had been completed, but prior to the submittal, a plant change was made in response to an Appendix R issue (i.e., hot shorts).

The plant change resulted in changing the normal position of the power-operated relief valve (PORV) block valves. Originally, the block valves were left in the open position and closed only in response to a leaking PORV. The modification changed the normal position of these valves to closed, with a signal to open on high reactor coolant system pressure.

This change was implemented via the change control process and resulted in a reduction in the consequence of a postulated piping failure. That is, the consequence changed from

a loss of coolant accident (LOCA) to a potential LOCA. The failure potential assessment was unchanged by this modification. As a result of this review, there were no changes to the number of inspections or inspection locations.

PLANT B

Plant B is an approved Class 1-only application that utilized the EPRI methodology. After the RI-ISI evaluations had been completed, but prior to the submittal, the PRA was updated.

The update consisted of additional initiating events (IE), revised initiating event frequencies, mitigating system modeling and data. An update to the RI-ISI evaluations was conducted to reflect the latest PRA information. As a result of this review, there were no changes to the consequence ranking and thus no changes to the risk ranking and element selection process.

PLANT C

Plant C is a Class 1-only application that utilized the Westinghouse Owners' Group (WOG) methodology. The following change was incorporated into the RI-ISI program update:

- Update to the PRA model.

No segments went from low safety significance (LSS) to high safety significance (HSS). The expert panel re-categorized six segments from HSS to LSS.
The net effect of the update on the number of examinations was that six VT-2 examinations were removed and one VT-2 examination was added for change in risk considerations.

PLANT D

Plant D is a Class 1 and Class 2 application that utilized the WOG methodology. The following changes were incorporated into the RI-ISI program update:

- Update to the PRA model
- Credit for operator action was removed from two segments.
- Consequences with operator action were revised for two segments removing the operator action and crediting automatic valve closure.
- Nine segments were identified as being potentially subjected to an active degradation mechanism and were included in an augmented program.
- The Structural Reliability and Risk Assessment (SRRA) failure probabilities of five segments were affected by various changes to the inputs.
- Test intervals for four segments were changed from 1.5 years to monthly.

The expert panel changed the categorization of 16 segments.

- Two segments were changed from LSS to HSS.
- The remaining 14 segments were changed from HSS to LSS.

Examination of one less segment was required to meet the change-in-risk criteria.
The net effect on the number of examinations included in the RI-ISI program was:

- Five new examinations were added,
- Eight previous examinations were removed, and
- Twelve VT-2 examinations were removed.

PLANT E

Plant E is a Class 1 and Class 2 application that utilized the WOG methodology. The following changes were incorporated into the RI-ISI program update:

- Update to the PRA model
- Credit for operator action was removed from two segments.
- Consequences with operator action were revised for two segments, removing the operator action and crediting automatic valve closure.
- Fourteen segments were identified as being potentially subject to an active degradation mechanism and were included in an augmented program.
- The SRRA failure probabilities of five segments were affected by various changes to the inputs.
- Test intervals for four segments were changed from 1.5 years to monthly.

The expert panel changed the categorization of 25 segments.

- Three segments were changed from LSS to HSS.
- The remaining 22 segments were changed from HSS to LSS.

The revised change-in-risk evaluation resulted in a few changes to meet the change-in-risk criteria.

- For one system, examination of one less segment was required.
- For another system, one additional segment was required to be examined, and the examinations associated with two segments had to be moved to other segments.

The net effect on the number of examinations was:

- Seven new examinations were added,
- Eight previous examinations were removed,
- Eight Flow Assisted Corrosion (FAC) examinations were no longer required per this program, but instead are part of an owner-defined program.
- Twenty VT-2 examinations were removed.

PLANT F

Plant F is a full-scope (i.e., Class 1, 2, and some three/NNS) application that utilized the EPRI methodology.

The update included reviewing a set of documents (e.g. engineering reports [ERs], NIS-2 reports, updated P-T sheets and industry activities in response to pipe cracking events; VC Summer, Oconee, TMI) issued between December 1998 and January 2003. Additional information sources used to develop the initial RI-ISI program also were evaluated (e.g. chemistry manual, operating procedures, insulation spec). These impacts and how they were dispositioned in this evaluation are described in more detail in Appendix A.

Based on this review, only a single ER impacted the RI-ISI program. The impact was limited to adding 13 welds to the program scope. As these welds were located in a low risk area (risk category 6), no additional inspections were required. In addition, as these welds were added via the change control process, they are identified to the ISI engineer via the existing change control process and would be captured regardless of RI-ISI update requirements.

In parallel with the review of plant changes, a review of the impact of the most recent PRA model on the RI-ISI evaluation was conducted. The current PRA has been updated to incorporate plant changes, the impact of the power uprate, as well as modeling enhancements. Details of this review are provided in Appendix A.

Although there were no changes in the PRA that impacted the RI-ISI program (i.e., consequence assignment, consequence rank, risk ranking or element selection, number of inspections), there was a change that had a minor impact on the change in risk assessment. The change in risk assessment used the conditional core damage probability (CCDP) value for large LOCAs as the bounding CCDP for risk categories with a high consequence (i.e., risk categories 1, 2 and 4). As part of the PRA update, CCDPs for LOCAs have changed. The impact of the updated CCDP values on the change-in-risk assessment showed that the existing program still meets acceptance criteria.

PLANT G

Plant G is a full-scope (i.e. Class 1, 2, and some three/NNS) application that utilized the EPRI methodology. Prior to the submittal, a review of plant changes was conducted as the RI-ISI evaluations had been conducted over a period of several years.

A review of plant documents (drawings, design changes, procedures) was conducted and identified one plant change that impacted the RI-ISI evaluations. This design change replaced stainless steel piping with carbon steel piping. This run of piping originally was identified as susceptible to intergranular stress corrosion cracking (IGSCC). Because of

the material change, IGSCC was no longer applicable. Thus, the RI-ISI evaluations and element selection were updated to reflect this change.

In parallel with the review of plant changes, a review of the impact of the most recent PRA model on the RI-ISI evaluations was conducted. The updated PRA was updated to incorporate plant changes and Boiling Water Reactor Owners' Group peer review, as well as modeling enhancements.

The RI-ISI evaluation for assessing the change in risk due to the RI-ISI program used the CCDP value for large LOCAs as the bounding CCDP for risk categories with a high consequence (i.e., risk categories 1, 2 and 4). As part of the PRA update, the bounding values—and therefore the delta risk results—did not change.

PLANT H

Plant H is a full-scope (i.e., Class 1, 2, 3, and some NNS) application that utilized a methodology based on that developed by the WOG. The following changes were incorporated into the RI-ISI program update:

- Update to the PSA model
- Initiation of hydrogen water chemistry and noble metals injection for IGSCC mitigation
- Power uprate.

Categorization of the segments based on the revised risk reduction worth (RRW)—the primary importance measure for the WOG methodology—from the risk evaluation resulted in the following:

- The number of segments designated HSS due to RRW reduced from 29 to 22

The expert panel had originally decided to classify any segment that would result in a large LOCA as HSS, regardless of the RRW of that segment, for defense-in-depth. The number of segments designated HSS for this reason increased from eight to 17. Because the number of inspections in the defense-in-depth segments was reduced, the total number of examinations was reduced from 85 to 66.

3.5 SUMMARY/CONCLUSIONS

This review provides a high degree of confidence that existing plant controls provide reasonable assurance that changes to plant level core damage frequency (CDF) due to RI-ISI programs updates will be insignificant. However, at a more detailed level, plant program and procedure changes were identified that could impact the RI-ISI program and its results (e.g., failure potential, consequence of failure, number of inspections).

The use of existing operating procedures, corrective action programs, and industry operating experience procedures in conjunction with PRA update information will be the cornerstone for the following recommended guideline practices.

4 GUIDELINES/RECOMMENDED PRACTICES

4.1 EXPEDITED REVIEWS

As discussed earlier, the number and location of inspections as determined by a RI-ISI program can be influenced by plant-specific and industry events. It is anticipated that plant and industry events impacting the postulated consequence of failure (e.g., PRA changes, expert panel deliberations) will not require RI-ISI programs to be updated in an expedited manner. It is not expected that these events would uncover shortcomings in the design basis or design basis assumptions. As such, these events are expected to result in a more informed state of knowledge rather than identify shortcomings in the RI-ISI inspection population. Thus, their inclusion would result in a refinement of the RI-ISI program rather than the need to conduct immediate examinations.

Conversely, it is possible that a new type of degradation might be uncovered or an existing type of degradation becomes more pronounced. Examples of this include the primary water stress corrosion cracking events at V.C. Summer and Tsuruga.

In response to these events, the industry has established a comprehensive plan to assess and make recommendations for industry action concerning this type of degradation mechanism. The interim recommendations have included increased inspection scope, personnel training and significant analysis and testing. Individual licensees, as part of their operating experience review programs, have done susceptibility reviews and conducted augmented inspections as they deemed appropriate.

Given the above, and the fact that other than augmented inspections (e.g., FAC, IGSCC in boiling water reactors) the remaining RI-ISI-identified inspections are conducted on a ten-year inspection interval, expedited updates of RI-ISI programs are not required.

4.2 PERIODIC REVIEWS

A RI-ISI program uses feedback of new relevant information to support the identification of high safety-significant piping locations. In addition to existing SXI practices, reviews should be performed on an ASME period basis and address the following areas.

Examination Results

The results of examinations performed should be reviewed for indications of leakage or flaws. This review contains two aspects. If the failure is due to a degradation mechanism postulated in the original RI-ISI evaluation, the inspection (type, frequency) specified for that degradation mechanism should be re-evaluated for adequacy. If the failure results from a degradation mechanism that is new, or different from that postulated in the original RI-ISI evaluation, all segments potentially affected by the new mechanism should be identified and the impact on failure potential evaluated. The risk-informed ISI program could be updated by either adding additional examination selections in accordance with the requirements for HSS piping structural elements in the identified segments, or by using the applicable portions of the same risk-informed selection process that originally established the risk-informed inspection program. This re-evaluation of the selections should be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to perform the entire risk-informed selection process again, but the evaluation for the changes to the piping selections that do occur would be documented.

Piping Failures

Potential piping failures can be identified by means other than NDE examinations performed per the RI-ISI program. Reviewing the plant corrective action program should identify plant-specific piping failures in areas other than those subject to the RI-ISI program. Monitoring via the operating experience review program should identify piping failures elsewhere in the industry. If a degradation mechanism is identified that is new or different from that postulated in the RI-ISI evaluation, all segments potentially affected by the new mechanism should be identified and the impact on failure potential evaluated. The risk-informed ISI program could be updated by either adding additional examination selections in accordance with the requirements for HSS piping structural elements in the identified segments, or by using the applicable portions of the same risk-informed selection process that originally established the risk-informed inspection program. This re-evaluation of the selections should be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to perform the entire risk-informed selection process again, but the evaluation for the changes to the piping selections that do occur would be documented.

PRA Update

Plant PRAs may be modified or updated over time. The decision to modify or update a PRA is licensee-specific, and currently there is no regulatory requirement for such revisions on any specific frequency. Future applications (e.g., proposed 10 CFR 50.69) may involve such requirements.

Generally, PRA changes fall into two categories. ASME standard RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Plants," Section 5, discusses PRA configuration control with respect to PRA maintenance and PRA upgrades.

PRA maintenance generally involves changes resulting from updated initiating event data, updated equipment performance data, modifications to plant equipment or procedures, etc. These changes may or may not result in significant impacts on PRA results (e.g., risk metrics, significant sequences, significant equipment, etc). The PRA organization within a plant is responsible for evaluating these issues on some periodicity and determining when they may result in the need for changes to the model or to assumptions impacting previously approved applications. Judgment is required, as the PRA model cannot practically be continuously adjusted to reflect all changes.

PRA upgrades involve changes to the methodology of the model (e.g., changing model platforms, changing initiating event grouping, making major changes to known significant areas such as reactor coolant pump seal leakage modeling for pressurized water reactors). As with PRA maintenance, PRA upgrades may or may not result in significant changes to the results, although they are generally more likely to do so than PRA maintenance.

As with other risk applications, PRA changes as discussed above should be evaluated for their impacts on RI-ISI. This evaluation should be performed by individuals knowledgeable in the PRA model and the RI-ISI methodology. The decision as to whether the RI-ISI methodology, or parts thereof, should be performed again, requires judgment relative to the magnitude of the overall changes to the PRA and taking into consideration the risk-significance of the RI-ISI application. It should be noted that many plants have a requirement to update the RI-ISI program on a ten-year interval (i.e., relief request is valid for one inspection interval), and PRA changes within that interval, depending on their magnitude, may not be risk-significant to incur interim changes to the program.

In many cases it should not be necessary to perform the entire risk-informed selection process again. For instance, as shown in appendix A for the EPRI RI-ISI methodology, as long as the consequence rank assignments are consistent between the original PRA and the updated PRA (e.g., initiating event CCDP and system unavailability stay within the allowable ranges), then these results can be documented and no further analysis is required. Table B-1 of Appendix B provides an example of such an approach.

To evaluate PRA changes for periodic updates (i.e., 40 months) for plants that have used the WOG RI-ISI methodology, the following process is suggested. The process is described using CDF as the example, but it applies equally to LERF. Identify the initiating events and systems modeled in the PSA portion of the RI-ISI program. Using results directly from the PSA model, compare the new plant total CDF, the individual initiating event's CDF, and system contribution CDF to the corresponding values (total plant, initiating event and system CDF) from the PSA model used for the latest complete RI-ISI

risk evaluation. For the individual initiating event and system contribution CDFs, only those initiating events and system contributions that are identified as consequences in the RI-ISI program are compared. If all of the new values are less than or equal to the values used for the latest complete RI-ISI risk evaluation, then no further evaluation is necessary for the periodic update. If there is an increase in any of the three (total plant, initiating event, or system CDF), then determine the percent increase with respect to that used for the RI-ISI program. If all of the increases are less than 25 percent then no further evaluation is necessary for the periodic update. If there is an increase greater than 25 percent then assess the impact by evaluating the new information at the appropriate level in the RI-ISI process. If there is no change to the Level 2 portion of the plant's PSA model and the total plant, initiating event, and system CDFs all remain the same or decrease, there is no need to evaluate LERF.

The evaluation cutoff of a 25 percent increase is based on the following considerations:

- The calculated segment RRWs includes an uncertainty analysis.
- The expert panel process incorporates both quantitative results and deterministic insights in classifying the segments as either high or low safety significant.
- A 25 percent increase in the CDF for a piping segment is required to increase the segment's RRW from 1.004 to 1.005.

If the PRA changes result in an increase in an initiating event CDF or system contribution and there are other changes affecting the RI-ISI program inputs (e.g., increase in failure probability), then the percent increase in the consequences should be factored into the evaluation of the other changes to determine the combined effect on the affected segments. Appendix B provides an example of such an approach.

Plant Design Changes

Plant design changes can be physical, programmatic or procedural. Physical changes can include new piping or equipment installation, or modification of existing equipment. These changes should be identified by the design control process and should be evaluated for impact on the scope of application, failure potential, consequence or segment definition. Programmatic changes can include such things as power uprating, change in fuel cycle, or implementation of plant chemistry changes. These changes should be identified by the design control process or by monitoring the licensing basis, and should be evaluated for impact on consequence evaluations or failure potential. Procedural changes can include modification to surveillance tests or operating procedures and can be identified by the procedure change review process. Procedural changes can affect consequence evaluations and failure potential. All segments potentially affected by any of these changes should be identified. The risk-informed ISI program could be updated either by adding additional examination selections in accordance with the requirements for HSS piping structural elements in the identified segments, or by using the applicable portions of

the same risk-informed selection process that originally established the risk-informed inspection program. This re-evaluation of the selections should be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to perform the entire risk-informed selection process again, but the evaluation for the changes to the piping selections that do occur would be documented.

Changes in Postulated Conditions

Many specific conditions are postulated and various assumptions are made during the RI-ISI evaluation process. The change control process and the corrective action process should be monitored for any changes to the conditions or assumptions. These can include such things as a change from salt water to freshwater in a system, assumption of valve leakage leading to thermal effects, assumption of complete isolation by a valve, or specific critical action times. All such changes should be evaluated for impact on failure potential and consequence evaluations. All segments potentially affected by any change should be identified. The risk-informed ISI program could be updated either by adding additional examination selections in accordance with the requirements for HSS piping structural elements in the identified segments, or by using the applicable portions of the same risk-informed selection process that originally established the risk-informed inspection program. This re-evaluation of the selections should be performed by inserting the new information at the appropriate level of the analysis. It may not be necessary to perform the entire risk-informed selection process again, but the evaluation for the changes to the piping selections that do occur would be documented.

Table 3-2 contains a summary of the changes to be evaluated for an update.

4.3 TEN-YEAR INTERVAL

Changes occurring as a result of periodic updates that were based on qualitative and/or quantitative re-evaluation results would be cumulatively evaluated for inclusion in the next subsequent inspection interval update. The subsequent inspection interval update includes a re-evaluation using the applicable portions of the same risk-informed selection process that originally established the risk-informed ISI program. This re-evaluation is performed by evaluating the new information at the appropriate step in the RI-ISI evaluation process. It may not be necessary to perform the entire risk-informed selection process again, but the evaluation for the changes to the piping selections that have occurred would be documented and include a change-in-risk evaluation. Consistent with existing SXI practices, the inspection interval update meets the requirements for the ISI program Edition and Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, IWA-2400, required to be used in accordance with 10CFR50.55a.

5 REPORTING

5.1 WITHIN THE CURRENTLY APPROVED INSPECTION INTERVAL

During an October 3, 2001, meeting between the NRC and the industry, it was agreed that the intent of the RI-ISI template process is to provide the NRC with the information necessary to conclude with reasonable assurance that the licensees:

- Conducted the RI-ISI evaluation consistent with a topical report and its safety evaluation (SE), and
- The change in risk as a result of the RI-ISI program is within acceptance criteria.

As such, the intent of the RI-ISI template process is to provide a fixed snapshot in time of the RI-ISI program and, therefore, the following may change without requiring NRC approval or notification:

- Delta risk numbers, provided they remain within acceptance criteria,
- Number of inspections, or
- Allocation of inspections.

NRC notification and approval would be required when:

- Changing from one methodology to another,
- Changing the scope of application (see note below), for example:
  - Class 1 only to Class 1 & 2,
  - Full scope to Class 1 only,
- Plant-specific impact of revised methodology on the SE (e.g. changes to the PWSCC temperature threshold in the EPRI methodology),
- Significant industry/plant event, not addressed by generic/methodology update (see section 4.1),
- ASME Section XI ten-year updates as required by plant-specific SE, or
- Changes that impacts the basis for NRC approval in the plant-specific SE are identified (e.g. plant specific commitment to meet NUREG-0313 versus EPRI BWRVIP-075 for IGSCC in BWRs).

Note: Minor changes to class boundaries (e.g., piping reroute, P&ID revisions) do not require re-submittal, as they do not impact the basis for the NRC's approval of the previous RI-ISI submittal.

It was agreed that generic conclusions discussed during the meeting apply to both "template" plants and "pilot" plants, unless there are other commitments in the plant's SE.

The above not withstanding, any licensee commitment not specifically addressed in the RI-ISI safety evaluation would be treated via existing plant commitment management processes.

5.2 SUBSEQUENT INSPECTION INTERVALS

5.2.1 Existing Process

Most plants that have received approval to implement a RI-ISI program have approval for a single inspection interval. As such, a new relief request is needed to continue the RI-ISI program into the next inspection interval.

The information necessary to support this new relief request consists of the following three items:

1. Identification of the number of welds deleted from the originally approved RI-ISI program,
2. Identification of the number of welds added to the proposed RI-ISI program for the new inspection interval as compared to the originally approved RI-ISI program, and
3. Confirmation that the change in risk assessment for the new inspection interval as compared to the last deterministic SXI inspection program meets the acceptance criteria of the original RI-ISI submittal.

Tables 5-1 and 5-2 present a recommended approach for providing the information discussed in items 1) and 2) above. Thus for subsequent intervals, a relief request should be prepared as part of the ten-year update that contains documentation (paragraph, letter) confirming that the change in risk assessment was conducted consisted with item 3) above and attaching a plant specific version of Table 5-1 or 5-2.

5.2.2 NRC Endorsement of ASME Nonmandatory Appendix

As of this writing, the ASME is processing a nonmandatory appendix that codifies the RI-ISI processes. Upon approval by ASME and endorsement by NRC in 10CFR50.55a, plants wishing to implement a RI-ISI program no longer would need prior NRC approval. (This assumes that the RI-ISI program is developed in accordance with the requirements of the ASME nonmandatory appendix and additional requirements imposed upon the appendix by the NRC).

6 REFERENCES

1. ASME Case N-560 “Alternative Examination Requirements for Class 1, Category B-J piping Welds,” Section XI Division 1, March 28, 2000.
2. ASME Case N-577 “Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method A,” Section XI Division 1, March 28, 2000.
3. ASME Case N-578 “Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B,” Section XI Division 1, March 28, 2000.
4. WCAP-14572, “Westinghouse Owners Group Application of Risk-Based Methods to Piping Inservice Inspection Topical Report,” Revision 1-NP-A, dated February 1999.
5. Electric Power Research Institute TR-112657, Rev B-A “Revised Risk-Informed Inservice Inspection Evaluation Procedure,” dated December 1999.
6. 10CFR50.55(a), Code of Federal Regulations, Domestic Licensing of Production and Utilization facilities, Codes and Standards.
7. ASME Boiler and Pressure Vessel Code, An International Code American Society of Mechanical Engineers, New York, New York.
8. U.S. NRC, “Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,” 51 Federal Register 30028, Aug. 21, 1986.
9. NUMARC 91-06, “ Guidelines for Industry Actions to Address Shutdown Management”
10. NEI 00-02, “Probabilistic Risk Assessment (PRA) Peer Review Process Guidance”
11. RA-S – 2002, “Probabilistic Risk Assessment for Nuclear Power Plant Applications,” 2002. American Society of Mechanical Engineers, New York, New York.

TABLE 3-1
RI-ISI Unique Inputs and References

INPUT DESCRIPTION (or equivalent)	ATTRIBUTE	INTENT	CONCLUSIONS
Shutdown Operating Procedures	Defines operating temperature at decay heat removal (DHR) initiation	Confirm operating temperatures and configurations do not change or remain consistent with RI-ISI evaluation. For example, DHR is not initiated until RCS is < 245F, the rate of steamline cooldown/heatup is maintained, removal of condensation continues, AFW, EFW and MFW flow rates and temperature differences are closely controlled.	Changes to operating procedures are conducted via 10CFR50.59 and provide reasonable assurance that they would not result in unacceptable increases in plant risk.
Normal System Operating Procedures	Defines start-up, shutdown, test and operating conditions	Confirm operating temperatures, pressures, flow rates, operating duration and configurations.	Changes to operating procedures are conducted via 10CFR50.59 and provide reasonable assurance that they would not result in unacceptable increases in plant risk.
Abnormal System Operating Procedures	Defines operator actions to mitigate consequences of a piping failure.	Confirm assumptions used in the consequence analysis.	Changes to operating procedures are conducted via 10CFR50.59 and provide reasonable assurance that they would not result in unacceptable increases in plant risk.

TABLE 3-1
RI-ISI Unique Inputs and References

INPUT DESCRIPTION (or equivalent)	ATTRIBUTE	INTENT	CONCLUSIONS
Loss of DHR	Defines operator actions during loss of DHR events	Confirm assumptions used in the consequence analysis of shutdown events.	Changes to operating procedures are conducted via 10CFR50.59 and provide reasonable assurance that they would not result in unacceptable increases in plant risk.
Shutdown Risk Management Plan	Defines plant-specific shutdown risk-management actions	Confirm shutdown risk-management program and activities are in place to manage risk.	Existing practices in accordance with 10CFR50.65(a)(4) meet this intent.
Repair/ Replacement Activities	Physical changes due to repair, replacement or modification activities	Confirm plant changes are reflected in the ISI and RI-ISI program.	Physical plant changes are implemented via the design change control process and therefore will continue to meet design basis requirements.
Maintain conformance with EPRI water chemistry	Control of contaminants and others species	Confirm failure potential assumptions, as applicable.	Most, if not all, plants are in compliance with these guidelines. It is highly unlikely that plants will implement changes that significantly degrade water quality.

TABLE 3-1
RI-ISI Unique Inputs and References

INPUT DESCRIPTION (or equivalent)	ATTRIBUTE	INTENT	CONCLUSIONS
Maintain use of hydrazine during plant start-up	Control of oxygen	Confirm failure potential assumptions, as applicable.	If a plant were to change operating practices, the change control process would require an evaluation to assure there are no significant negative effects (e.g., component reliability).
Industry events	Postulated degradation mechanisms	Confirm applicability of various degradation mechanisms to particular system and operating configurations (e.g., temperature, material).	See Section 3.
Insulation spec	External chloride stress corrosion cracking (ECSCC)	Confirm that the installed insulation material is in compliance with Reg Guide 1.36, or equivalent.	If a plant were to change the insulation spec, the change control process would require an evaluation to assure there are no significant negative effects (e.g., component reliability).

TABLE 3-1
RI-ISI Unique Inputs and References

INPUT DESCRIPTION (or equivalent)	ATTRIBUTE	INTENT	CONCLUSIONS
Transient cycle counting program	Thermal fatigue (TF)	Confirm that assumption in TF evaluation remains valid or that cycle counting program adequately monitors, evaluates and identifies appropriate actions.	Given the transition to license renewal, this is even less of an issue. In addition, industry work on thermal fatigue and environmentally assisted fatigue will be provided to the industry once complete.
Operator action times	Credited in the consequence analysis	Captured by PRA update or identified and reviewed as part of the RI-ISI update.	See Section 3.
Tech spec surveillance test intervals (STIs)	Used in consequence ranking effort	Monitor tech spec changes	See Section 3.

TABLE 3-2
Disposition Table for Periodic RI-ISI Evaluation

Change	Examples	Programmatic Disposition	Technical Disposition
Examination results	• Leakage, flaw	• Existing Section XI requirements • Existing augmented program requirements	• If new or different degradation mechanism - Determine affected segments - Evaluate impact on failure potential • If postulated degradation mechanism, revisit degradation mechanism evaluation (Re: inspection frequency)
Piping failures (plant-specific)	• Failure due to new degradation mechanism; failure occurs in systems not previously susceptible to type of failure. • If postulated degradation mechanism, revisit degradation mechanism evaluation (Re: inspection frequency)	• Existing Section XI requirements, • Existing corrective action program	• If new or different degradation mechanism - Determine affected segments - Evaluate impact on failure potential • If postulated degradation mechanism, revisit degradation mechanism evaluation (Re: inspection frequency)
Piping failures (industry)	• Failure due to new degradation mechanism; failure occurs in systems not previously susceptible to type of failure. • If postulated degradation mechanism, revisit degradation mechanism evaluation (Re: inspection frequency)	• Operating experience program • Existing industry guidance or regulatory directives	• If new or different degradation mechanism - Determine any affected segments - Evaluate impact on failure potential • If postulated degradation mechanism, revisit degradation mechanism evaluation (Re: inspection frequency)
PRA update	• New initiating event • New system function • More detailed model • Initiating event and failure data • Revised success criteria	• Additional evaluation per recommendation in Section 4	• Evaluate impact on consequence evaluation • Evaluate impact on scope of application

TABLE 3-2
Disposition Table for Periodic RI-ISI Evaluation

Change	Examples	Programmatic Disposition	Technical Disposition
Plant design change (physical)	• Snubber / support change	• Existing design control process	• Evaluate impact on failure potential, if applicable
	• New piping or equipment installation	• Existing design control process	• Evaluate impact on scope of application, failure potential, consequence, segment definition
Plant design change (programmatic)	• Power uprating / station blackout diesel / 24 month fuel cycle	• Existing design change control process	• Evaluate impact on failure potential and consequence evaluations
	• Water chemistry change	• Existing design control process or licensing basis	• Evaluate impact on failure potential
Plant design change (procedural)	• Pump test change from quarterly to refueling in standby system	• Additional evaluation per recommendation in Section 4	• Evaluate impact on failure potential, consequence, risk ranking
	• EOP, AOP, NOP, SAGMs	• Additional evaluation, per recommendation in Section 4	• Evaluate impact on failure potential and consequence evaluations
Change in Postulated Conditions or Assumptions (See Table 3-1)	• Change from salt water to fresh water • Check valve leaking or not leaking • Critical action times	• Existing corrective Action program, • Existing change control process	• Evaluate impact on failure potential and consequence evaluations

TABLE 3-3
PRA Changes by Technical Element
(Per ASME PRA Standard)

Technical Element	Potential Change	Example PRA Change	Cause of Change	Impact on RI-ISI
IE (initiating event)	Increase in frequency	Small LOCA (SLOCA)	Reflect more recent industry data	Systems used to respond to a SLOCA could become more important
		Loss of PCC train A (LPCCA)	Reflect plant data, modeling changes, peer review	Systems used to respond to a LPCCA could become more important
	Decrease in frequency	Small LOCA (SLOCA)	Reflect more recent industry data	Systems used to respond to a SLOCA could become less important
		Loss of PCC train A (LPCCA)	Reflect plant data, modeling changes, peer review	Systems used to respond to a LPCCA could become less important
	New IE	Partition LOCA sizes	More refined modeling	Only if relevant to pressure boundary failures within the scope of the RI-ISI applications its relevance to RI-ISI piping pressure boundary
		Loss of DC bus versus train	More discrete modeling	Only if relevant to pressure boundary failures within the scope of the RI-ISI application
	Delete IE	Combine RT and TT into NPT	Quantification speed	None
		Eliminate steamline break outside containment	Low risk contributor	Loss of basis for consequence assignment
Accident sequence analysis (AS)	Should be covered by the other elements (e.g. sucess criteria, system analysis)			
Success Criteria (SC)	New SC is more conservative	3 of 4 steam generators (S/G) now required for secondary heat removal	S/G tube plugging	EFW less reliable, therefore EFW subsystems and other SHR paths (e.g. F&B) more important

TABLE 3-3
PRA Changes by Technical Element
(Per ASME PRA Standard)

Technical Element	Potential Change	Example PRA Change	Cause of Change	Impact on RI-ISI
Success Criteria (SC)	New SC is more conservative	3 of 4 steam generators (S/G) now required for secondary heat removal	S/G tube plugging	EFW less reliable, therefore EFW subsystems and other SHR paths (e.g. F&B) more important
	New SC is less conservative	2 of 4 ECCS injection paths now required	Updated MAAP analyses	ECCS more reliable, therefore individual ECCS subsystems less important
Systems Analysis (SA)	System determined to be more reliable	HPSI (injection phase)	MOV failure rate reduced	ECCS, injection phase more reliable, but no impact on recirculation phase, therefore no change.
		PCC dual pump contribution	Common cause modeling updated	PCC more reliable, therefore individual trains less important
	System determined to be less reliable	EDG fail to run	Incorporation of plant specific data	LOSP more important, therefore systems in RI-ISI scope that are important for mitigating LOSP could become more important
		ESFAS actuation	More accurate modeling	Spatial effect may or may not be more important.
Human Reliability Analysis (HR)	New recovery action	Refill of reactor water storage tank (RWST) during LOCA scenario	EOPs updated	More options for responding to small LOCAs, therefore ECCS less important, more credit for SHR (EFW)
	Less reliable operator action	Switchover to recirculation	Less time available due to revised level set points caused by vortex concerns	All postulated breaks requiring switchover become more important.
		Isolation during midloop	Less time available due to power uprate	Impact on assumed shutdown risk contribution
Data Analysis	Should be covered by the other elements (e.g. initiating event frequency, systems analysis)			
Internal	New	New IE	Piping routing change/addition	For piping within the RI-ISI scope, new analysis/evaluation required

TABLE 3-3
PRA Changes by Technical Element
(Per ASME PRA Standard)

Technical Element	Potential Change	Example PRA Change	Cause of Change	Impact on RI-ISI
flooding (IF), including indirect effects	flooding source	Increased existing IE frequency	Piping routing change/addition	For piping within the RI-ISI scope, new analysis/evaluation required
	Elimination of a flooding source	Deletion of IE	Piping routing change/deletion	For piping within the RI-ISI scope, new analysis/evaluation required
		Reduced IE frequency	Piping routing change/addition	For piping within the RI-ISI scope, new analysis/evaluation required
	New flooding targets	Updated internal flooding analysis	Re-location of instrument racks	Could increase the importance of some previously analyzed piping failures
	Removal of flooding targets	Updated internal flooding analysis	Equipment relocated or raised above flood height	Could decrease the importance of some previously analyzed piping failures
	Barriers strengthened	Updated internal flooding analysis	Industrial door upgraded to water tight	Could decrease the importance of some previously analyzed piping (e.g. better separation/isolation)
				Could increase the importance of some previously analyzed piping (e.g. industrial door was assumed to fail preventing flood build-up).
Quantification (QU)	Truncation	More sequences included	Better computer hardware/software	Should not be an issue
	Results	CDE,RRW values	Different software code	Should not be an issue.
LERF Analysis	Should be covered by the other elements as discussed above (e.g. IE, success criteria, and system analysis).			

Abbreviations:

SLOCA – small break LOCA
PCC – primary component cooling water
DC – direct current
RT –reactor trip
TT – turbine trip
NPT – normal plant trip
ECCS – emergency core cooling system
HPSI – high pressure safety injection
EDG – emergency diesel generator
MOV – motor operated valve
MAAP – modular accident analysis program
LOSP – loss of offsite power
EFW – emergency feedwater
F&B – feed and bleed
SHR – secondary heat removal
ESFAS – engineered safety features actuation system
EOPs – emergency operating procedures
HEP – human error probability
CDF – core damage frequency
RRW – risk reduction worth

Table 5-1
Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1st Aprvd RI-ISI Interval		New RI-ISI Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RPV	2 (2)	High (High)	High	TT, (IGSCC)	Medium (Medium)	B-F	1	1[3]		1[3]	
RPV	2 (2)	High (High)	High	CC, (IGSCC)	Medium (Medium)	B-F	12	3[4]		3[4]	
RPV	2	High	High	CC	Medium	B-J	10	3		3	
RPV	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	6	1[5]		1[5]	
RPV	4	Medium	High	None	Low	B-J	8	1		1	
BB	4	Medium	High	None	Low	B-J	134	14		14	
BB	6a	Low	Medium	None	Low	B-F	1	0		0	
						B-J	49	0		0	
BB	7a	Low	Low	None	Low	B-J	2	0		0	
BG	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	5	1		1	
BG	4	Medium	High	None	Low	B-F	2	0		0	
						B-J	111	12		12	
BG	7a	Low	Low	None	Low	B-J	12	0		0	
BD	6a	Low	Medium	None	Low	C-F-2	42	0		0	
FC	5a (3)	Medium (High)	Medium	TT, (FAC)	Medium (High)	C-F-2	3	1		1	
FC	5a	Medium	Medium	TT	Medium	C-F-2	16	2		2	
FC	6a (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	8	0		0	
FC	6a	Low	Medium	None	Low	B-F	2	0		0	
						B-J	20	0		0	
						C-F-2	17	0		0	
FC	7a	Low	Low	None	Low	B-J	2	0		0	

Table 5-1
Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1st Aprvd RI-ISI Interval		New RI-ISI Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
BC	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	1	1[6]		1[6]	
BC	4	Medium	High	None	Low	B-F	3	2		2	
						B-J	110	13		13	
						C-F-2	28	0		0	
BC	5a	Medium	Medium	TASCS	Medium	C-F-2	3	1		1	
BC	6a	Low	Medium	None	Low	B-J	6	0		0	
						C-F-2	479	0		0	
BC	7a	Low	Low	None	Low	B-J	37	0		0	
						C-F-2	208	0		0	
BE	2	High	High	TT	Medium	B-J	3	1		1	
BE	4	Medium	High	None	Low	B-J	32	7		7	
						C-F-2	41	1		1	
BE	6a	Low	Medium	None	Low	B-J	9	0		0	
BE	7a (5b)	Low (Medium)	Low	None (FAC)	Low (High)	C-F-2	4	0		0	
BE	7a	Low	Low	None	Low	B-J	2	0		0	
						C-F-2	188	0		0	
BJ	4	Medium	High	None	Low	C-F-2	12	2		2	
BJ	5a	Medium	Medium	TT	Medium	C-F-2	4	1		1	
BJ	6a	Low	Low	None	Low	C-F-2	85	0		0	
FD	4	Medium	High	None	Low	B-F	2	0		0	
						B-J	15	2		2	
FD	5a	Medium	Medium	TT	Medium	C-F-2	22	3		3	
FD	6a	Low	Medium	None	Low	B-J	3	0		0	
						C-F-2	56	0		0	

Table 5-1

Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	1st Aprvd RI-ISI Interval		New RI-ISI Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
AB	4	Medium	High	None	Low	B-J	102	11		11	
AB	6a (3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	5	0		0	
AB	6a	Low	Medium	None	Low	B-J	166	0		0	
						C-F-2	37	0		0	
AE	2 (1)	High (High)	High	TASCS, TT, (FAC)	Medium (High)	B-J	3	2		2	
AE	2 (1)	High (High)	High	TASCS, (FAC)	Medium (High)	B-J	7	1		1	
AE	2 (1)	High (High)	High	TT, (FAC)	Medium (High)	B-J	2	0		0	
AE	2	High	High	TASCS, TT	Medium	B-J	10	3		3	
AE	2	High	High	TASCS	Medium	B-J	18	5		5	
AE	2	High	High	TT	Medium	B-J	4	0		0	
AE	4 (1)	Medium (High)	High	None (FAC)	Low (High)	B-J	23	3		3	
AE	4	Medium	High	None	Low	B-J	14	2		2	
AE	5a	Medium	Medium	TASCS, TT	Medium	B-J	4	1		1	
						C-F-2	1	0		0	
AE	5a	Medium	Medium	TASCS	Medium	B-J	4	1		1	
						C-F-2	1	0		0	
AE	5a	Medium	Medium	TT	Medium	C-F-2	5	0		0	
AE	6a (3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	1	0		0	
AE	6a	Low	Medium	None	Low	C-F-2	12	0		0	
BF	7a	Low	Low	None	Low	C-F-2	20	0		0	
BH	4	Medium	High	None	Low	B-J	16	2		2	
BH	6a	Low	Medium	None	Low	B-J	21	0		0	
BH	7a	Low	Low	None	Low	B-J	5	0		0	
AP	6a	Low	Medium	None	Low	C-F-2	6	0		0	

Notes for Table 5-1

1. Systems are described in Table 3.1
2. The column labeled "Other" is generally used to identify plant augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows plant augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. As stated in Section 3.5 of this template, the RI-ISI program achieved a 9.3% sampling without relying on plant augmented inspection program locations beyond those selected for RI-ISI purposes either due to the presence of other damage mechanisms, or to satisfy Risk Category 4 selection requirements. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
3. This piping weld has been selected for examination per the augmented inspection program for IGSCC (Category "C") and for RI-ISI purposes due to the presence of other damage mechanisms.
4. These three piping welds have been selected for examination per the augmented inspection program for IGSCC (two Category "C" and one Category "E") and for RI-ISI purposes due to the presence of other damage mechanisms.
5. This piping weld has been selected for examination per the augmented inspection program for IGSCC (Category "C") and is being credited for RI-ISI purposes.
6. This piping weld has been selected for examination per the augmented inspection program for IGSCC (Category "C") and is being credited for RI-ISI purposes.

Table 5-2

INSPECTION LOCATION SELECTION COMPARISON TO ASME SECTION XI and WCAP-14572 Rev. 1-NP-A

System	Number of HSS Segments (No. of HSS in Augmented Program / Total No. of Segments in Augmented Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count[i]		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		1st Approved RI-ISI Interval		New RI-ISI Interval[a]	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations	SES Matrix Region	Number of Exam Locations
ACC	0	VF	Class 1	B-J	36	0	9	0	-	0	-	0
AFW[c]	11 (5 / 16)	Corrosion	Class 2	C-F-2	80	~50	6	3	1A, 1B	5	1A, 1B	5
			Class 3				0	0		3+3[e]		3+3[e]
AS	2	TF	Non-Code				0	0	1A, 1B	2	1A, 1B	2
BD[c]	6 (6 / 12)	VF, FAC	Class 2	C-F-2	54	0	0	0	1A, 1B	3	1A, 1B	3
			Non-Code				0	0		3		3
CC	6	TF, VF	Class 3				0	0	1A, 1B, 2	13+4[e]	1A, 1B, 2	13+4[e]
CH	8 (0 / 3)	TF, VF,SCC	Class 1	B-J	156	~60	39	6	1A, 1B, 2	12+6[b]+4[e]	1A, 1B, 2	12+6[b]+4[e]
			Class 2	C-F-1	10	~20	0	0		1+3[e]		1+3[e]
CN[c]	0 (0 / 6)	Wastage	N/A				0	0	-	0	-	0
CS	0 (0 / 2)	Wastage, SCC	Class 2	C-F-1	120	0	9	0	-	2[g]	-	2[g]
CW[d]	4	Wastage	N/A				0	0	-	0	-	0

Table 5-2

INSPECTION LOCATION SELECTION COMPARISON TO ASME SECTION XI and WCAP-14572 Rev. 1-NP-A

System	Number of HSS Segments (No. of HSS in Augmented Program / Total No. of Segments in Augmented Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count[i]		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		1st Approved RI-ISI Interval		New RI-ISI Interval[a]	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations	SES Matrix Region	Number of Exam Locations
ECC	7 (0 / 1)	Stratification	Class 1	B-J	16	0	4	0	1A, 1B, 2	12	1A, 1B, 2	12
			Class 2	C-F-1	320	0	24	0	2	1	2	1
EE	0	Wastage/ Corrosion	N/A				0	0	-	0	-	0
FC	0	TF, VF, SCC	N/A				0	0	-	0	-	0
FWc	13 (13 / 17)	Wastage, TF	Class 2	C-F-2	80	0	6	0	1A, 1B	0	1A, 1B	0
			Non-Code				0	0		7		7
HHIc	14 (1 / 5)	TF, VF, SCC	Class 2	C-F-2	450	0	63	0	1A, 1B, 2	15+2g	1A, 1B, 2	15+2g
LHIc	7 (1 / 1)	TF, VF, SCC	Class 2	C-F-2	305	~20	23	4	1A, 1B, 2	7+3b+2g	1A, 1B, 2	7+3b+2g
MSc	3 (3 / 23)	Wastage, TF	Class 2	C-F-2	240	0	18	0	1A, 1B	2+1f	1A, 1B	2+1f
RC	11	TF, VF, Strip/Strat, SCC	Class 1	B-F	18	0	18	0	1A, 1B, 2	9	1A, 1B, 2	9
				B-J	584	~50	146	12		11+10g, h +3b		11+10g,h +3b
RH	4	SCC, VF	Class 1	B-J	16	0	4	0	1A, 1B, 2	1	1A, 1B, 2	1

Table 5-2

INSPECTION LOCATION SELECTION COMPARISON TO ASME SECTION XI and WCAP-14572 Rev. 1-NP-A

System	Number of HSS Segments (No. of HSS in Augmented Program / Total No. of Segments in Augmented Program)	Degradation Mechanism(s)	Class	ASME Code Category	Weld Count[i]		ASME XI Examination Methods (Volumetric (Vol) and Surface (Sur))		1st Approved RI-ISI Interval		New RI-ISI Interval[a]	
					Butt	Socket	Vol & Sur	Sur Only	SES Matrix Region	Number of Exam Locations	SES Matrix Region	Number of Exam Locations
			Class 2	C-F-1	160	0	12	0		4		4
RS	2	TF, VF, SCC	Class 2	C-F-1	54	0	4	0	1A, 1B	2	1A, 1B	2
SWd	8	TF	Class 3				0	0	1A, 1B	5+3e	1A, 1B	5+3e
VS	2	TF, VF	Class 3				0	0	1A, 1B, 2	2	1A, 1B, 2	2
TOTAL	108 (29 / 89)		Class 1	B-F	18	0	18	0		9 NDE		9 NDE
				B-J	808	~110	202	18		46 NDE + 13 VIS		46 NDE +13 VIS
			Class 2	C-F-1	664	~20	49	0		10 NDE + 3 VIS		10 NDE + 3 VIS
				C-F-2	1209	~90	116	7		36 NDE + 4 VIS		36 NDE + 4 VIS
			Class 3			0	0		23 NDE + 10 VIS	23 NDE + 10 VIS		
			Non-Code			0	0		12 NDE	12 NDE		
			Total	2699	~220	385	25	136 NDE + 30 VIS	136 NDE + 30 VIS	136 NDE + 30 VIS		

Summary: There are no changes to the number of exams in the RI-ISI program for the new ten-year interval. *Or if there were changes.* Five exams were added (or removed) in the RI-ISI program for the new ten-year interval. Prior ASME Section XI selects a total of 385 non-destructive exams while the proposed RI-ISI program selects a total of 136 exams (166 - 30 visual exams), which results in a 65% reduction.

Degradation Mechanisms: FAC – Flow-Assisted Corrosion, SCC – Stress Corrosion Cracking; Strip/Strat – Striping/Stratification; VF – Vibratory Fatigue; TF – Thermal Fatigue;

Notes for Table 5-2

a. System pressure test requirements and VT-2 visual examinations shall continue to be performed in all ASME Code Class 1, 2, and 3 systems.
b. VT-2 area exam at specific location.
c. Augmented programs for erosion-corrosion and/or high energy line break continue.
d. Pipe coatings program will be maintained.
e. VT-2 for entire segment.
f. Segment MS-34 has no weld; VT-2 for entire segment.
g. Ten examinations added for change in risk considerations.
h. Six examinations added for defense-in-depth at the reactor vessel outlet nozzle to pipe welds.
j. Section XI does not require NDE weld examination of Class 3 welds. The number of welds in Class 3 systems is not known.

APPENDIX A

Details of the Example Plant Applications

PLANT A

Plant A is a Class 1 only application that utilized the EPRI methodology. This application was submitted to the NRC in 2002. After the RI-ISI evaluations had been completed but prior to the submittal, a plant change was made in response to an Appendix R issue (i.e., hot shorts).

The plant change resulted in changing the normal position of the power operated relief valve (PORV) block valves. Originally, the block valves were left in the open position and closed only in response to leaking PORVs. The modification changed the normal position of these valves to closed with a signal to open on high RCS pressure.

This change was implemented via the change control process and resulted in a reduction in the consequence of a postulated piping failure. That is, the consequence changed from a LOCA to a potential LOCA. The failure potential assessment was unchanged by this modification. As a result of this review, there were no changes to the element selection or change in risk assessments.

PLANT B

Plant B is a Class 1 only application that utilized the EPRI methodology. This application was submitted to the NRC in February, 2002 and approved in February 2003. After the RI-ISI evaluations had been completed but prior to the submittal, the PRA had been updated.

The update consisted of additional initiating events, revised initiating event frequencies, mitigating system modeling and data. An update to the RI-ISI evaluations was conducted to reflect the latest PRA information. As a result of this review, there were no changes to the consequence ranking and thus no changes to the risk ranking and element selection process.

PLANT C

Plant C is a Class 1 only application that utilized the Westinghouse Owners Group (WOG) methodology. This application was approved by the NRC in January 2001 and inspections began during the Spring 2002 outage.

As part of the interval update, a review of plant documents (drawings, design changes, procedures) and the plant PRA model was conducted. The review identified no changes that would impact the RI-ISI program except for an update to the PRA model.

Significant changes to the PRA model included:

- Updates to common cause failure (CCF) basic events. Many changes involved separating common cause fail-to-run from fail-to-start as well as combinations of 2 of 3 or 2 of 4.
- Human error probability basic events were updated.

- Initiating events in the model were updated with the latest industry data. Several changed significantly.
- The SLOCA frequency was revised.
- The seal LOCA terms were updated to reflect completed installation of the high temperature RCP seals, which provide a substantial risk benefit.
- Unavailability data were updated to include the average unavailabilities from the last three years.
- Component reliability data were updated to include the reliabilities from the last three years.
- Event trees were updated to model LOCA's or tube ruptures in any of the RCS legs.
- The seal LOCA model event trees were updated to include a new function for the core uncovered event and the seal LOCA does not develop event.
- Flooding was added to the internal events PRA.
- Several additional CCF models were generated.
- Recovery screening values were developed.
- Fail-to-start and fail-to-run common cause failure terms were developed for the component cooling pumps.
- Numerous changes were made to the disallowed maintenance fault tree.
- Various minor fault trees changes were performed.

Categorization of the segments based on the risk reduction worth from the risk evaluation resulted in the following:

- No segment RRW went from less than 1.001 or between 1.005 and 1.001 to greater than 1.005.
- One segment RRW went from greater than 1.005 to between 1.005 and 1.001.
- Thirteen segment RRWs went from between 1.005 and 1.001 to less then 1.001.

In general, the updated component reliability resulted in lower calculated values for CDF and LERF. The overall result for the segment whose RRW changed from greater than 1.005 to between 1.005 and 1.001 reflected a reduction in the small LOCA contribution sufficient to change the numerical categorization.

The expert panel agreed with the updated numerical results and the associated assumptions. The panel decided that the previous expert panel decisions would be maintained except for six segments on the emergency core cooling (ECC) system. The one segment whose RRW went from greater than 1.005 to between 1.005 and 1.001 was maintained HSS to keep the examination percentages unchanged (UT) per the original NRC submittal. The six segments on the ECC system were 2 inch nominal pipe size and ¾ inch nominal pipe size lines that currently received a VT-2 examination. The segments have RRWs less than 1.001 (base case), but the original expert panel categorized them HSS due to concerns associated with a single check valve separation between the LSS segment and an adjoining HSS segment. The new expert panel reviewed results from another plant expert panel on the same lines (determined to be LSS by the other expert

panel), looked at the current testing methodology of the valves in question, and ultimately reclassified the segments LSS. They noted the check valves are tested each time the valve is flow balance tested, eliminating a stuck open valve concern.

It should be noted that three six-inch segments in the same area were kept HSS to maintain the ultrasonic examination percentages and to address defense-in-depth. Two VT-2 examinations were added to the reactor coolant system to meet the change-in-risk criteria.

The net effect of the update on the number of examinations was that six VT-2 examinations were removed and two VT-2 examinations were added.

PLANT D

Plant D is a Class 1 and Class 2 application that utilized the WOG methodology. This application was approved by the NRC in October 2001 and inspections began during the Fall 2001 outage.

As part of the periodic update, a review of plant documents (drawings, design changes, procedures) and PRA model was conducted and identified the following changes in addition to an update to the PRA model.

- Credit for operator action was removed from 2 segments.
- Consequences with operator action were revised for 2 segments removing the operator action and crediting automatic valve closure.
- Nine segments were identified as being potentially subjected to an active degradation mechanism and were included in an augmented program.
- The SRRA failure probabilities of 5 segments were affected by various changes to the inputs.
- Test intervals for 4 segments were changed from 1.5 years to monthly.

The effect of the above changes on the risk evaluation results was as follows:

- No segment RRWs went from less than 1.001 to greater than 1.005.
- Five segment RRWs went from between 1.005 and 1.001 to greater than 1.005. One segment RRWs increased due to an increase in the SRRA failure probability. The piping LERF values for the remaining four segments decreased but the total plant LERF decreased by a greater amount. This coupled with the uncertainty analysis resulted in the RRWs increasing for these four segments. All five segments had previously been categorized as HSS by the expert panel.
- No segment RRWs went from greater than 1.005 to less than 1.001.
- Twenty segment RRWs went from greater than 1.005 to between 1.005 and 1.001. All segment RRWs decreased due to a decrease in the PRA values. Two segment RRWs also decreased due to an increase in the test interval frequency. Of these twenty segments only one segment was recategorized LSS by the expert panel.

The expert panel changed the categorization of 16 segments.

- Two segments were changed from LSS to HSS based on removal of credit for an operator action that resulted in the with operator action RRWs changing from less than 1.001 to between 1.005 and 1.001. The without operator RRWs for these segments remained between 1.005 and 1.001.
- The remaining 14 segments were changed from HSS to LSS.
    - Seven of these segments changed based on changes from the PRA update. The RRWs for one segment went from greater than 1.005 to between 1.005 and 1.001. The RRWs for one segment went from between 1.005 and 1.001 to less than 1.001. The RRWs for the remaining five segments remained between 1.005 and 1.001 but were lower.
    - Seven segments had previously been conservatively ranked HSS even though the final original quantitative results supported LSS and there were no deterministic reasons for ranking these segments HSS. During the update, the expert panel decided to recategorize the segments based on the quantitative results. The RRWs for these seven segments remained basically the same. The RRWs for four of the segments were less than 1.001 while the RRWs for the other three segments were between 1.005 and 1.001.

For the change-in-risk evaluation, one less segment was required to meet the change-in-risk criteria.

The net effect on the number of examinations included in the RI-ISI program was:

- Five new examinations were added,
- Eight previous examinations were removed, and
- Twelve VT-2 examinations were removed.

No changes were made for defense-in-depth.

PLANT E

Plant E is a Class 1 and Class 2 application that utilized the WOG methodology. This application was approved by the NRC in October 2001 and inspections began during the Spring 2002 outage.

As part of the periodic update, a review of plant documents (drawings, design changes, procedures) and PRA model was conducted and identified the following changes in addition to an update to the PRA model.

- Credit for operator action was removed from two segments.
- Consequences with operator action were revised for two segments removing the operator action and crediting automatic valve closure.
- Fourteen segments were identified as being potentially subjected to an active degradation mechanism and were included in an augmented program.
- The SRRA failure probabilities of five segments were affected by various changes to the inputs.

- Test intervals for four segments were changed from 1.5 years to monthly.

The effect of the above changes on the risk evaluation was as follows:

- No segment RRWs went from less than 1.001 to greater than 1.005.
- Nine segment RRWs went from between 1.005 and 1.001 to greater than 1.005. Three segment RRWs increased due to an increase in the SRRA failure probability. The piping LERF values for the remaining six segments decreased but the total plant LERF decreased by a greater amount. This coupled with the uncertainty analysis resulted in the RRWs increasing for these six segments. All but one of these nine segments who RRWs went from between 1.005 and 1.001 to greater than 1.005 had previously been categorized HSS by the expert panel. The ninth segment was recategorized by the expert panel to HSS based on a potential active degradation mechanism that increased the SRRA failure probability and RRWs.
- No segment RRWs went from greater than 1.005 to less than 1.001.
- Sixteen segment RRWs went from greater than 1.005 to between 1.005 and 1.001. All segment RRWs decreased due to a decrease in the PRA values. Two segment RRWs also decreased due to an increase in the test interval frequency. Of these 16 segments, six segments were recategorized LSS by the expert panel.

The expert panel changed the categorization of 25 segments.

- Three segments were changed from LSS to HSS. Two of these segments changed based on removal of credit for operator action that resulted in the with operator action RRWs changing from less than 1.001 to between 1.005 and 1.001. The without operator action RRWs for these two segments remained between 1.005 and 1.001. The third segment changed from LSS to HSS based on a potential active degradation mechanism that increased the SRRA failure probability and RRWs.
- The remaining 22 segments were changed from HSS to LSS.
    - Fifteen of these segments changed based on changes from the PRA update. The RRWs for six segments went from greater than 1.005 to between 1.005 and 1.001. The RRWs for one segment went from between 1.005 and 1.001 to less than 1.001. The RRWs for the remaining 8 segments remained between 1.005 and 1.001 but were lower.
    - Seven segments had previously been conservatively ranked HSS even though the final original quantitative results supported LSS and there were no deterministic reasons for ranking these segments HSS. During the update, the expert panel decided to rank the segments based on the quantitative results. The RRWs for these seven segments remained basically the same. The RRWS for four of the segments were less than 1.001 while the RRWs for the other three segments were between 1.005 and 1.001.

The revised change-in-risk evaluation resulted in a few changes to meet the change-in-risk criteria.

- For one system one less segment was required to be examined.
- For another system, one additional segment was required to be examined and the examinations associated with two segments had to be moved to other segments.

The net effect on the number of examinations was:

- Seven new examinations were added,
- Eight previous examinations were removed,
- Eight FAC examinations were no longer required per this program but instead are part of an owner-defined program.
- 20 VT-2 examinations were removed.

No changes were made for defense-in-depth.

PLANT F

Plant F is a full scope (i.e., Class 1, 2, and some 3/NNS) application that utilized the EPRI methodology. This application was approved by the NRC in December 1998 and inspections began during the January 1999 outage.

The effort included reviewing a set of documents (e.g. engineering reports [ERs], NIS-2 reports, updated Pressure-Temperature sheets, and industry activities in response to pipe cracking events; VC Summer, Oconee, TMI) issued between December 1998 and January 2003. In addition, additional information sources used to develop the initial RI-ISI program were also evaluated (e.g. chemistry manual, operating procedures, insulation spec).

Disciplines involved in this review included staff familiar with the ISI program, system engineering, repair/replacement, plant operations, design, power uprate, steam generator replacement, and PRA. Based on this review, only a single ER impacted the RI-ISI program. The impact was limited to adding 13 welds to the program scope. As these welds were located in a low risk area (risk category 6), no additional inspections were required. In addition, as these welds were added via the change control process, they are identified to the ISI engineer via the existing change control process and would be captured regardless of RI-ISI update requirements.

In parallel with the review of plant changes, a review of the impact of the most recent PRA model on the RI-ISI evaluation was conducted. The current PRA has been updated to incorporate plant changes, the impact of the power uprate as well as modeling enhancements.

PRA changes can impact the RI-ISI evaluation in the following ways:

- success criteria
- initiating event conditional core damage probability (CCDP)
- mitigative system unavailability (i.e., equivalent trainworth)

- combination events (i.e., postulated break causes an initiating event and impacts mitigative equipment)
- containment performance
- change in risk evaluation.

The following summarizes the impact on the RI-ISI program from the PRA changes.

Success Criteria:

There were no changes made to the PRA, as to the success criteria used in the RI-ISI evaluations.

Pipe Break That Results in Initiating Events Only:

There are a number changes reflected in the most recent PRA as compared to the PRA used for the RI-ISI evaluations. These include changes to CCDP values, new initiating events (e.g. loss of additional AC buses), as well as modified initiating events (e.g. steam/feed line breaks on SG-A inside MSIVs). The end result of this review is that the existing RI-ISI program can remain unchanged.

Pipe Breaks That Impact Mitigating Systems:

The update included a comparison of the "equivalent trainworths" based upon system unavailabilities used in the RI-ISI evaluations as compared to the current PRA. The end result of this review is that the existing RI-ISI program can remain unchanged.

Pipe Break That Results in Combination Events:

There were no changes made to the PRA with regards to combination events relative to their impact on the RI-ISI program.

Containment Performance:

There were no changes made to the PRA, as to its impact on containment performance with respect to the RI-ISI program.

Change In Risk Evaluation:

The RI-ISI evaluation for assessing the change in risk due to the RI-ISI program used the CCDP value for large LOCAs as the bounding CCDP for risk categories with a high consequence (i.e., risk categories 1, 2 and 4). As part of the PRA update, CCDPs for LOCAs have changed. The impact of the updated CCDP values on the change-in-risk assessment showed that the existing program still meets acceptance criteria.

PLANT G

Plant G is a full scope (i.e., Class 1, 2, and some 3/NNS) application that utilized the EPRI methodology. This application was submitted in October 1999 and approved by the NRC in September 2000. Prior to the submittal, a review of plant changes was conducted as the RI-ISI evaluations had been conducted over a period of time.

A review of plant documents (drawings, design changes, procedures) was conducted and identified one plant change that impacted the RI-ISI evaluations. This design change replaced stainless steel piping with carbon steel piping. This run of piping was originally identified as susceptible to IGSCC. Because of the material change, IGSCC was no longer applicable. Thus, the RI-ISI evaluations and element selection were updated to reflect this change.

In parallel with the review of plant changes, a review of the impact of the most recent PRA model on the RI-ISI evaluations was conducted. The PRA had been updated to incorporate plant changes, peer review by the BWR operators Group as well as modeling enhancements.

PRA changes can impact the RI-ISI evaluation in the following ways:

- success criteria
- initiating event conditional core damage probability (CCDP)
- mitigative system unavailability (i.e., equivalent trainworth)
- combination events (i.e., postulated break causes an initiating event and impacts mitigative equipment)
- containment performance
- change in risk evaluation.

As to the impact on the RI-ISI evaluations, the following summarizes the findings:

Success Criteria:

There were no changes made to the PRA, as to the success criteria used in the RI-ISI evaluations.

Pipe Break That Results in Initiating Events Only:

There were a number changes reflected in the most recent PRA as compared to the PRA used for the RI-ISI evaluations. These include changes to CCDP values, additional/modified initiating events as well as changes to truncation levels. These changes did not impact the results of the RI-ISI evaluations.

Pipe Breaks That Impact Mitigating Systems:

All changes to the system unavailabilities used in the RI-ISI evaluations remained within the consequence ranks as used in the RI-ISI evaluations.

Pipe Break That Results in Combination Events:

There were no changes made to the PRA with regards to combination events relative to their impact on the RI-ISI program.

Containment Performance:

There were no changes made to the PRA, as to its impact on containment performance with respect to the RI-ISI program.

Change In Risk Evaluation:

The RI-ISI evaluation for assessing the change in risk due to the RI-ISI program used the CCDP value for large LOCAs as the bounding CCDP for risk categories with a high consequence (i.e., risk categories 1, 2 and 4). As part of the PRA update the bounding values, and therefore the delta risk results, did not change.

PLANT H

Plant H is a full scope (i.e., Class 1, 2, 3, and some NNS) application that utilized a methodology based on that developed by WOG. This application was submitted in May 2000 and was approved by the NRC in January 2001.

The following changes were incorporated into the RI-ISI program update:

- update to the PSA model
- initiation of hydrogen water chemistry and noble metals injection for IGSCC mitigation
- power uprate

In accordance with the original submittal and the NRC safety evaluation report, this new information was evaluated to determine if the RI-ISI methodology and/or inspection program had changed as a result. It was decided that it would be a prudent use of resources to also include the additional parameters to be evaluated for a periodic update review:

- plant design feature changes
- plant procedure changes
- equipment performance changes
- examination results
- individual plant and industry failure information
- corrective action program

A major revision to the plant PRA was performed for the following reasons:

- incorporate more plant-specific data
- incorporate plant design changes
- incorporate latest NRC data regarding initiating events, common cause, and RPS failure
- resolve issues identified in the Peer Review

- provide a more detailed Level 2 (LERF) analysis

The impact of these changes was as follows:

- The Core Damage Frequency (CDF) attributable to each of the six dominant systems (MS, FW, RWCU, RECIRC, RHR, and CS) changed by less than 1%.
- No new High Safety Significant (HSS) segments were identified

The CDF and LERF were re-calculated for the Base case, the original Section XI examinations, the Augmented examinations, and the Risk-Informed examinations. In each case the CDF was reduced by approximately an order of magnitude, and the LERF was reduced by approximately two orders of magnitude. The RI program based on the new PSA still resulted in a net risk reduction.

The Refueling Outage NIS-1 and NIS-2 Owners Summary Reports were reviewed. No change to the RI-ISI program was indicated by this review.

System Engineers performed a review to identify plant equipment performance changes and industry and plant piping failures. They specifically addressed:

- Plant design feature changes
- Plant procedure or Surveillance Instruction changes
- Component or associated equipment performance changes
- Individual plant and/or industry failure information
- System or component trending data

Programmatic plant design feature changes identified by system engineers that had impact on the RI-ISI program included:

- Power uprate
- Initiation of Hydrogen Water Chemistry and Noble Metals Injection as mitigative actions for IGSCC.

New failure rates were calculated due to changes in stress calculations from the power uprate. New failure rate values were also calculated to include the effect of HWC/NMI.

Categorization of the segments based on the revised risk reduction worth, which is the primary importance measure for the WOG methodology, from the risk evaluation resulted in the following:

- The number of segments designated HSS due to RRW reduced from 29 to 22

The expert panel had originally decided to classify any segment that would result in a large LOCA as HSS, regardless of the RRW of that segment, for defense-in-depth. The number of segments designated HSS for this reason increased from 8 to 17.

Due to the reduced number of inspections in the defense-in-depth segments, the total number of examinations was reduced from 85 to 66.

Appendix B

Supplemental Information

Table B-1
EPRI Consequence Category Assignment for Pipe Failures Causing Initiating Events

	Base Case Results				Updated Results			
Initiating Event	IEF (Events/yr)	CDF (Events/yr)	CCDP (CDF/IEF)	Consequence	IEF (Events/yr)	CDF (Events/yr)	CCDP (CDF/IEF)	Consequence
T1 - Turbine trip	7.6E-01	2.3E-06	<1E-06	Low	2.39E-01	1.01E-07	4.22E-07	Low
T2 – Loss of PCS	2.5E-01	9.0E-07	3.6E-06	Medium	8.73E-02	8.54E-07	9.78E-06	Medium
T3 – LOSP	5.8E-02	1.7E-06	2.9E-05	Medium	3.16E-02	4.22E-07	1.33E-05	Medium
T4 - Excessive FW	9.4E-04	1.9E-09	<1E-06	Low	9.40E-04	1.94E-10	2.06E-07	Low
T5 - Steam/Feed break	1.1E-03	1.1E-09	1.0E-06	Medium				
T5A – Line Break on SG-A Inside MSIV					5.50E-04	4.18E-08	7.59E-05	Medium
T5B – Line Break on SG-B Inside MSIV					5.50E-04	4.18E-08	7.60E-05	Medium
T5C – Steam Line Break Outside MSIV					2.40E-03	1.79E-08	7.46E-06	Medium
T6 - Reactor Trip	2.0E+00	6.0E-06	<1E-06	Low	9.24E-01	4.34E-07	4.69E-07	Low
T7 – Loss of SW	5.5E-03	2.1E-06	3.8E-04	High	1.80E-03	1.34E-07	7.43E-05	Medium
T8 – Loss of SW P4A	7.4E-02	2.1E-07	2.8E-06	Medium	1.38E-01	1.23E-07	8.94E-07	Low
T9 – Loss of SW P4B	7.4E-02	2.0E-07	2.7E-06	Medium	1.38E-01	2.19E-07	1.59E-06	Medium
T10 - Loss of DC D01	3.9E-04	9.8E-06	2.5E-02	High	3.94E-04	4.53E-08	1.15E-04	High
T11 - Loss of DC D02	3.9E-04	1.1E-06	2.8E-03	High	3.94E-04	2.34E-08	5.94E-05	Medium
T12 - Loss of AC A3	3.9E-04	3.2E-06	8.2E-03	High	3.94E-04	1.68E-08	4.26E-05	Medium
T13 - Loss of AC A4					3.94E-04	1.43E-10	3.62E-07	Low
T14 - Loss of AC B5					1.04E-03	1.96E-07	1.89E-04	High
T15 - Loss of AC B6					1.04E-03	1.86E-08	1.79E-05	Medium
T16 – Spurious					4.59E-03	3.35E-08	7.29E-06	Medium

Table B-1
EPRI Consequence Category Assignment for Pipe Failures Causing Initiating Events

	Base Case Results				Updated Results			
MSIS								
T17 – Closure MSIVs					3.97E-02	4.21E-07	1.06E-05	Medium
T18 – Loss of Condenser Vacuum					7.80E-02	7.54E-07	9.67E-06	Medium
S – Small LOCA	5.0E-03	1.7E-06	3.4E-04	High	2.95E-03	1.52E-06	5.16E-04	High
M - Medium LOCA	1.0E-03	1.7E-06	1.7E-03	High	6.60E-05	1.66E-07	2.51E-03	High
A – Large LOCA	1.0E-04	1.4E-06	1.4E-02	High	6.79E-05	2.25E-07	3.32E-03	High
R – SGTR	9.8E-03	9.5E-08	9.7E-06	Medium				
RA – SGTR on SG-A					3.50E-03	5.15E-08	1.47E-05	Medium
RB – SGTR on SG-B					3.50E-03	5.15E-08	1.47E-05	Medium
RVR – Vessel Rupture					2.70E-07	2.70E-07	1.00E+00	High
ISLOCA				High				
Initiating Event	IEF (Events/yr)	CDF (Events/yr)	CCDP (CDF/IEF)	Consequence	IEF (Events/yr)	CDF (Events/yr)	CCDP (CDF/IEF)	Consequence
T1 - Turbine trip	7.6E-01	2.3E-06	<1E-06	Low	2.39E-01	1.01E-07	4.22E-07	Low
T2 – Loss of PCS	2.5E-01	9.0E-07	3.6E-06	Medium	8.73E-02	8.54E-07	9.78E-06	Medium
T3 – LOSP	5.8E-02	1.7E-06	2.9E-05	Medium	3.16E-02	4.22E-07	1.33E-05	Medium
T4 - Excessive FW	9.4E-04	1.9E-09	<1E-06	Low	9.40E-04	1.94E-10	2.06E-07	Low
T5 - Steam/Feed break	1.1E-03	1.1E-09	1.0E-06	Medium				
T5A – Line Break on SG-A Inside MSIV					5.50E-04	4.18E-08	7.59E-05	Medium
T5B – Line Break on SG-B Inside MSIV					5.50E-04	4.18E-08	7.60E-05	Medium
T5C – Steam Line Break Outside MSIV					2.40E-03	1.79E-08	7.46E-06	Medium

Table B-1
EPRI Consequence Category Assignment for Pipe Failures Causing Initiating Events

	Base Case Results				Updated Results			
T6 - Reactor Trip	2.0E+00	6.0E-06	<1E-06	Low	9.24E-01	4.34E-07	4.69E-07	Low
T7 – Loss of SW	5.5E-03	2.1E-06	3.8E-04	High	1.80E-03	1.34E-07	7.43E-05	Medium
T8 – Loss of SW P4A	7.4E-02	2.1E-07	2.8E-06	Medium	1.38E-01	1.23E-07	8.94E-07	Low
T9 – Loss of SW P4B	7.4E-02	2.0E-07	2.7E-06	Medium	1.38E-01	2.19E-07	1.59E-06	Medium
T10 - Loss of DC D01	3.9E-04	9.8E-06	2.5E-02	High	3.94E-04	4.53E-08	1.15E-04	High
T11 - Loss of DC D02	3.9E-04	1.1E-06	2.8E-03	High	3.94E-04	2.34E-08	5.94E-05	Medium
T12 - Loss of AC A3	3.9E-04	3.2E-06	8.2E-03	High	3.94E-04	1.68E-08	4.26E-05	Medium
T13 - Loss of AC A4					3.94E-04	1.43E-10	3.62E-07	Low
T14 - Loss of AC B5					1.04E-03	1.96E-07	1.89E-04	High
T15 - Loss of AC B6					1.04E-03	1.86E-08	1.79E-05	Medium
T16 – Spurious MSIS					4.59E-03	3.35E-08	7.29E-06	Medium
T17 – Closure MSIVs					3.97E-02	4.21E-07	1.06E-05	Medium
T18 – Loss of Condenser Vacuum					7.80E-02	7.54E-07	9.67E-06	Medium
S – Small LOCA	5.0E-03	1.7E-06	3.4E-04	High	2.95E-03	1.52E-06	5.16E-04	High
M - Medium LOCA	1.0E-03	1.7E-06	1.7E-03	High	6.60E-05	1.66E-07	2.51E-03	High
A – Large LOCA	1.0E-04	1.4E-06	1.4E-02	High	6.79E-05	2.25E-07	3.32E-03	High
R – SGTR	9.8E-03	9.5E-08	9.7E-06	Medium				
RA – SGTR on SG-A					3.50E-03	5.15E-08	1.47E-05	Medium
RB – SGTR on SG-B					3.50E-03	5.15E-08	1.47E-05	Medium
RVR – Vessel Rupture					2.70E-07	2.70E-07	1.00E+00	High
ISLOCA				High				

Example of Factoring PRA Changes with Other Changes That Affect a WOG RI-ISI Program

For this example, the comparison of the CDF from the PRA model used for the last complete risk evaluation to the current PSA model resulted in the following:

- The total plant CDF decreased.
- The CDF for the initiating events and systems modeled in the PSA portion of the RI-ISI program remained the same or decreased with the exception of the SLOCA and MLOCA initiating events. The changes in the CDF for these IEs are presented below.

Initiating Event	CDF from PRA Model Used for Last Complete Risk Evaluation	CDF from Latest PRA Model	Percent Increase
SLOCA	1.62E-06	1.86E-06	15%
MLOCA	4.75E-08	5.23E-08	10%

Since the increase for SLCOA and the MLOCA IE are less than 25% no additional evaluation is required based on the PRA. However, one segment, RC-020 had an increase in the SRRA failure probability. Taking into account only the increase in the failure probability, the segment piping CDF without operator action increased to 1.04E-07.

The consequences associated with failure of this segment include a MLOCA and SLOCA. Since the CDF for these initiating events increased, the effect of the increases should be factored into the segment piping CDF as follows:

Segment piping CDF = 1.04E-07 * (1.86E-06 / 1.62E-06) * (5.23E-08 / 4.75E-08) = 1.31E-07

As an alternative to avoid some potential over conservatism, the IE factors could be applied to the individual pressure boundary CDFs and then the total summed. In this example, the pressure boundary CDFs including the revised failure probabilities are 5.69E-08 and 4.70E-08 for the SLOCA and MLOCA respectively. Thus the equation would be:

Segment piping CDF = (5.69E-08 * 1.86E-06 / 1.62E-06) + (4.70E-08 * 5.23E-08 / 4.75E-08) = 1.17E-07

To estimate the RRWs associated with the changes for this segment, the following steps are taken. To estimate the without operator action CDF RRW, a segment with the same or slightly higher without operator action CDF is identified from the last complete risk evaluation spreadsheets. This other segment without operator action CDF RRW is used to estimate the revised RRW. The same process is repeated for CDF with operation action and LERF without and with operator action as appropriate.

Note because this is an estimate of the RRW, strong consideration should be given to making segments with estimated RRWs of 1.004 or higher HSS.