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Materials Reliablity Program: Probabilistic Risk Assessment of Alloy 82/182 Piping Butt Welds (MRP-116NP)

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Final Report, October 2004

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REPORT SUMMARY

This report provides results of probabilistic fracture mechanics analyses to assess the probability of leaks, the probability of rupture due to crack growth, and the change in core damage frequency (CDF) resulting from PWSCC of Alloy 82/182 butt welds.

Background

In early October 2000, the V. C. Summer Plant shut down for a normal refueling outage and conducted a walk-down to search for boron deposits. During the walk-down, significant boron deposits were discovered in the vicinity of the reactor vessel Loop A outlet nozzle to pipe weld. Leakage records showed that leakage from all sources was well below the plant technical specification limit of 1.0 gpm. Ultrasonic tests performed on the pipe from the outside surface were inconclusive, but ultrasonic tests performed from the inside surface revealed a single axial flaw in the weld near the top of the pipe. Supplemental eddy current testing revealed several other possible anomalies, some of which were later confirmed to be flaws. Since that time, flaws have been discovered in a number of other Alloy 182 butt welds, and it may be anticipated that others will be found in the future.

Objectives

To analyze the likelihood and consequences of leakage and/or failure of dissimilar metal piping butt welds due to primary water stress corrosion cracking (PWSCC).

Approach

The project team has completed a probabilistic evaluation of Alloy 82/182 butt welds in pressurized water reactor (PWR) primary systems in Westinghouse, Combustion Engineering (CE), and Babcock and Wilcox (B&W) designed plants. The evaluation was built on basic input information used for deterministic structural integrity evaluations already performed and was aimed at providing a more realistic view of the margins that exist in these systems.

The project team has developed probability density function descriptions (that is, uncertainties) for all key input parameters needed for a butt weld fracture analysis. Examples include initiated flaw characteristics, such as depth and length; stress levels (operating, residual, and cyclic stresses); crack initiation time; crack growth rate (CGR), PWSCC, and fatigue; and accuracy of in-service inspections (ISIs). This information was used as input to probabilistic fracture mechanics (PFMs) models already in existence that have been used previously for PWSCC in head penetration nozzles and fatigue crack growth (FCG) in piping. Using the uncertainties and PFM models, the team calculated and benchmarked probabilities of observed flaws or leaks with limited cracking experience.

Using results from the already completed deterministic fracture analyses for the butt weld configurations in the Westinghouse, CE, and B&W nuclear steam supply system (NSSS) designs, the team determined limiting weld configurations, locations, and operating conditions. As part of this task, characteristics of the limiting transients (such as temperature, pressure, stress, and frequency) that are postulated to occur at the limiting locations in each design were identified. Several specific analyses were carried out for the B&W designs since there are differences with the CE design, even though both have carbon steel primary-loop piping.

The work on PFM model input development was coordinated with other work being performed by EPRI's Material Reliability Project (MRP) Butt Weld Working Group. Specifically, butt weld residual stress work was used in estimating stresses from repairs and benchmarking model-to-field failures, such as for the Virgil C. Summer hot leg nozzle weld leak. The team performed probabilistic fracture mechanics analyses for the limiting butt welds in large diameter pipes and several smaller diameter pipes for the Westinghouse, CE, and B&W NSSS designs. Sensitivity studies also were performed for many key input parameters. One of the more important parameters is for residual stress due to weld repairs during fabrication. Effects of ISI (accuracy and frequency) also were quantified probabilistically.

Consequences of butt weld failure, in terms of potential for core damage, and any associated uncertainties were obtained from either of these two sources: (1) probabilistic risk assessment (PRA) results for the limiting plants already determined in the previously completed deterministic evaluations and (2) representative consequences from typical plants that have performed a risk-informed evaluation for their piping ISI program. The project team combined these consequences and failure probabilities to determine potential risk in terms of the mean values of core damage, including effects of uncertainties in both probabilities and consequences.

Results

Results showed that the current American Society of Mechanical Engineers (ASME) Code Section XI ISI requirements for a service life of 40 years is adequate from a risk perspective.

EPRI Perspective

Due to recent PWSCC events of Alloy 82/182 butt welds, the industry, acting through EPRI's Materials Reliability Program, developed an interim safety assessment report and continued work on a final safety assessment to assure continued safe operation of these plants. This work is used as input to the final safety assessment for Alloy 82/182 pipe butt welds. As this work was underway, new information was discovered that could affect the calculations documented here. However, since the safety assessment will form the basis for recommended visual and nondestructive examinations, the industry opted to use the calculations as developed to the original workscope. The existing analyses and future analyses will be used as inputs into the Inspection and Evaluation Guidelines under development.

Keywords

PFM, PRA, PWSCC, Alloy 600, Alloy 82/182, Butt Welds, Risk-Informed, Probabilistic Fracture Mechanics, Probabilistic Risk Assessment Primary, Water Stress Corrosion Cracking

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1
INTRODUCTION

Cracking in Alloy 82/182 butt welds was not observed in operating PWR plants until the year 2000, when several incidents occurred. The first cracking event was found in June of that year, in Ringhals Unit 3, but the indications were thought to be shallow artifacts, and the plant was allowed to remain in service without repair. The second was in the reactor vessel outlet nozzle to safe end weld of Ringhals Unit 4, in July of 2000. Several small axial cracks were found and removed with a boat sample by electrical discharge machining (EDM). In both cases, the welds were Alloy 182, and the cracks were axial.

The next major incident occurred in October 2000, when the Virgil C. Summer plant was found to have a through-wall flaw in the same region as the Ringhals plants–the reactor vessel outlet nozzle to safe end weld. In this case, the weld was a field weld that had experienced multiple repairs during the construction process. In early October 2000, the Virgil C. Summer plant was shut down for a normal refueling outage, and a walkdown was undertaken to search for boric acid deposits, as is done to begin each outage. During the walkdown, significant boron deposits were discovered in the general vicinity of, and directly on, the Loop A outlet nozzle to pipe weld. Insulation was removed, and leakage-monitoring records were searched. Leakage records showed a nearly constant value of 0.3 gpm unidentified leakage from all sources. This is well below the plant Technical Specification limit of 1.0 gpm.

Ultrasonic inspection (UT) performed on the pipe from the inside surface initially revealed a single axial flaw near the top of the pipe. Followup exams conducted in the spring of 2002 revealed that there were several flaws, all but one of which were axial, and that the largest axial flaw was a through-wall flaw. The flawed region was removed, and a new spool piece welded in place, restoring this region to its original condition. The Virgil C. Summer outlet nozzle to pipe weld was repaired with Alloy 52, for a portion of the thickness exposed to the reactor coolant, and the remainder of the weld was filled with Alloy 82.

In November 2002, a surface indication was reported at Tihange Unit 2 in the Alloy 182 weld region of the pressurizer hot leg surge nozzle to safe end weld. The UT inspection technique was automatic, performed by AIB Vincottes of Belgium. The indication was axially oriented, and confined entirely to the weld, which had been repaired during construction and was not stress relieved. The depth of the indication was reported as 0.157-0.315 inches and the length was 1.02 inches. The indication was not repaired, and reinspection of the region in April 2003 showed that the indication had not changed in size or character.

In 2003, flaws were found during ISI of the Alloy 182 safe end regions of a pressurizer safety and the relief nozzles of Tsuruga Unit 2 in Japan. Axial flaws were found in each of the two

nozzles with one flaw in the relief valve nozzle being through wall. These nozzles were repaired with spool-piece replacements.

In October 2003, a part through-wall flaw was found in the Alloy 182 weld between the surge line and the hot leg nozzle at Three Mile Island (TMI) Unit 1, a B&W-designed plant. Again, the flaw was axially oriented, and was repaired by a structural weld overlay.

With this increasing population of cracks in Alloy 82/182 welds, and the likelihood of additional cracking occurring in the future, it is important to assess the safety and integrity of these weld regions. Deterministic evaluations have already been completed for these regions in Westinghouse, CE, and B&W designs. The goal of the work reported here is to use PFM techniques to assess the risk to plant safety from potential cracking in these regions to evaluate impacts on leak-before-break (LBB) assessments, and to determine if changes are needed in the ASME Section XI ISI requirements.

2
BACKGROUND FOR THE RISK ASSESSMENT

Deterministic stress and fracture evaluations were completed for all the Alloy 82/182 butt weld locations in the Westinghouse, CE, and B&W designed plants. For each location, the piping loads were compiled for all operating plants. In addition, using deterministic methods, the limiting cases were evaluated to determine the size of the flaw that could lead to piping failure, and the leak rate as a function of through-wall flaw size was calculated. The deterministic results showed that the margin between the flaw that gives detectable leakage and the critical flaw size is very large for large diameter pipes and decreases as pipe diameter decreases.

The significance of the safety case for longitudinal flaws is reduced, because the length of the flaw is limited to the width of the susceptible weld material. Therefore, for the maximum flaw length (~2.5 inches), the burst pressures were calculated. They were found to be 5.7 ksi or greater, which is well above the maximum operating pressure of 2.25 ksi. Therefore, large margins exist for longitudinal flaws. Service experience has shown that axial flaws are much more likely to occur than circumferential flaws.

While less likely to occur, circumferential flaws are not self limiting, and so detailed deterministic evaluations were carried out for each location. In each case, the flaw size necessary to cause a detectable leak, and then the time necessary for such a flaw to propagate to a critical length, were determined. These deterministic evaluations used a number of worst-case assumptions. These assumptions included bounding loads and residual stresses, as well as conservative crack growth predictions. The results showed that there were large margins for the large-diameter pipes, and smaller margins for the smallest-diameter pipes.

The deterministic results led to a concern that the inspection frequencies now being employed by Section XI might be in need of revision for the smaller pipe sizes. Therefore, it was decided to perform a series of risk studies for these pipes using PFM and PRA methods to assess concerns. The results of this evaluation are summarized in this report.

3 PFM METHODOLOGY

Probabilistic fracture mechanics models are used to calculate the small- and large-leak (100–1500 gpm) probabilities for Alloy 82/182 butt welds in piping nozzles. The time to initiation and CGR of the initiated flaw due to PWSCC are calculated with the same PFM models that were used previously for calculating the probability of failure in the base metal of Alloy 600 vessel head penetration nozzles [2]. The PFM models for stress intensity factor (SIF) of axial and circumferential flaws, crack growth due to fatigue, and the benefits of ISI are the same as those used previously for piping risk-informed ISI (RI-ISI) [3]. The methodology employed in these PFM models was reviewed and found to be acceptable by the NRC [4,5,6]. The RI-ISI PFM models also satisfy the requirements for computer codes for estimating failure probabilities in NUREG 1661 [7] Chapter 4, "Probabilistic Structural Mechanics." Highlights of critical parameters in regards to these PFM models and the associated input information for the probabilistic analyses, which is summarized in Table 3-1, are described in the following subsections.

Table 3-1
Summary of Information Used for PFM Analysis of Alloy 82/182 Butt Welds

Input Information for PFM Models	Source of Information	Ref. No.
Butt Weld Outside Diameter (inch)	Vendor Calculations	8, 9, and 10
Butt Weld Wall Thickness (inch)	Vendor Calculations	8, 9, and 10
Butt Weld Operating Temperature (°F)	Vendor Calculations	8, 9, and 10
Butt Weld Max. Residual Stress (ksi)	Alloy 600 Head Pen. Models and EPRI-MRP Residual Stress Study	2 and 10
Butt Weld Pressure Hoop Stress (ksi)	Vendor Calculations	8, 9, and 10
Deadweight and Thermal Stress (ksi)	Vendor Calculations	8, 9, and 10
Hoop/Axial Fatigue Stress Range (ksi)	Vendor Calculations	9 and 10
Fatigue Stress Ratio (min/max)	Vendor Calculations	9 and 10
Year Number for First Inspection (ISI)	Structural Reliability and Risk Assessment (SRRA) for Pipe RI-ISI	3 and 11
Frequency for Subsequent ISIs (years)	SRRA for Pipe RI-ISI	3 and 11
Cumulative Effect of ISI (1=No 2=Yes)	SRRA for Pipe RI-ISI	3 and 11

Depth at 50% Probability of Nondetection (PND)	SRRA for Pipe RI-ISI	3 and 11
PND Exponential Slope With Crack Depth	SRRA for Pipe RI-ISI	3 and 11
Hours at Temperature per Year	Alloy 600 Head Pen. Models	2
PWSCC Initiation Time Coefficient (hrs)	Alloy 600 Head Pen. Models	2
Combined Factors on Time to Initiation	Benchmarking with V.C. Summer	8
Initiation Activation Energy (cal/mole)	Alloy 600 Head Pen. Models	2
Stress Exponent for Initiation Time	Alloy 600 Head Pen. Models	2
Crack Depth at Initiation (inch)	Alloy 600 Head Pen. Models	2
Crack Length at Initiation (inch)	Alloy 600 Head Pen. Models	2
PWSCC Growth Rate Coefficient (in/hr)	EPRI-MRP CGR Expert Panel	12
Threshold Stress Intensity (ksi-in^.5)	EPRI-MRP CGR Expert Panel	12
PWSCC Rate Activation Energy (cal/mole)	EPRI-MRP CGR Expert Panel	12
PWSCC Rate Stress Intensity Exponent	EPRI-MRP CGR Expert Panel	12
Weld FCG Factor on Alloy 600 in Air	ANL Corrosion Fatigue Report	13
FCG Stress Intensity Range Exponent	ANL Corrosion Fatigue Report	13
Number of Fatigue Cycles per Year	Vendor Calculations	9 and 10
Limit Crack Depth for Small Leak (inch)	SRRA for Pipe RI-ISI	3 and 11
Flow Stress for Full Weld Break (ksi)	SRRA for Pipe RI-ISI	3 and 11
Hoop/Axial Stress Evaluation Condition (ksi)	Vendor Calculations	8, 9, and 10
Crack Length Limit for Through-Wall Flaw	Vendor Calculations	8, 9, and 10
Min. Detectable Leak Crack Length (inch)	SRRA for Pipe RI-ISI	3 and 11
Evaluation Condition Frequency per Year	Vendor Calculations	8, 9, and 10

3.1 Time to Initiation

Because of the very limited data on initiation of Alloy 82/182 butt weld material due to PWSCC, the currently available PWSCC initiation models [2,3] for the Alloy 600 base metal are used with an input adjustment factor to account for the differences in behavior between weld metal and base metal. This adjustment factor is used in the PFM models to account for:

- Differences between Alloy 600 base metal and Alloy 82/182 weld metal
- Effects of weld microstructure and microchemistry
- Effects of weld surface finish and cold work level

The value of this combined adjustment factor is determined using the benchmarking process, which is described in Section 4, to give a reasonable probability of the through-wall axial flaw that was found in Virgil C. Summer at the time the vessel outlet nozzle weld leak was discovered (17 years) [8]. The size of the PWSCC initiated flaw of .059 inches deep by .354 inches long [2] in Alloy 82/182 weld metal is the same as that for Alloy 600 base metal due to the lack of data to use in this regard.

3.2 PWSCC Crack Growth

Since the existing PFM models for reactor vessel head penetrations [2] already included crack growth due to PWSCC, the new correlation for weld metal (Alloy 82, 132, and 182) developed by the MRP PWSCC Crack Growth Expert Panel [12] can be used directly. Only the input parameters for the original Alloy 600 model needed to be updated.

The PWSCC CGR of 2.47E-07 $K^{1.6}$ inch/hour at 325°C (617°F), which is shown in Figure 3-1, is used as specified by the MRP PWSCC Crack Growth Expert Panel [12]. The value for the crack growth coefficient is calculated using the same activation energy that was used for the Alloy 600 base metal. The threshold SIF is set to approximately zero and the exponent is set to 1.6. Since most bi-metallic butt welds contain both alloys (weld metal and buttering), the coefficient value for only Alloy 182 is conservatively used because the Alloy 82 value would be a factor of 2.6 lower, as shown in Figure 3-1 [12].

3.3 Fatigue Crack Growth

Existing PFM models for FCG for piping RI-ISI [3] are used directly for the PFM models for butt welds. Figures 14(a) and 14(b) of the Argonne National Laboratory (ANL) Report [13], indicate that the FCG of Alloy 82/182, respectively, are best characterized as a factor on the Alloy 600 CGR in air. Analyses of this data are used to develop the mean value and uncertainty for the factor on weld FCG as discussed in the Section 4, Uncertainties. The effects of stress ratio R and temperature on the Alloy 600 CGR in air in Section 3.1 of the NUREG Report, CR-5864, [11] are also added to the PFM models for butt welds.

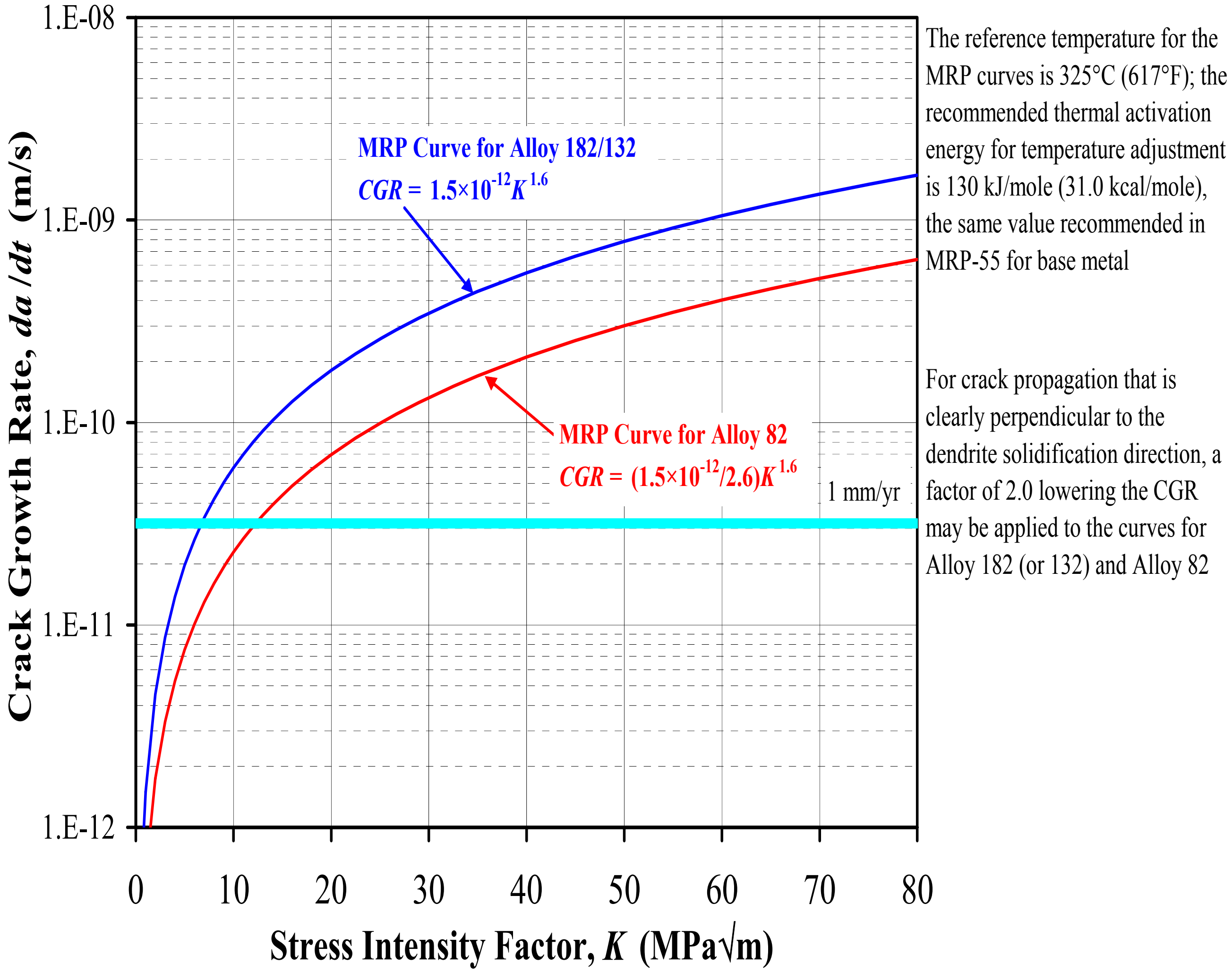

Figure 3-1
PWSCC Crack Growth Curves from Alloy 600 MRP Expert Panel [12]

Since most bi-metallic butt welds contain both alloys (weld metal and buttering), the median value for only Alloy 182 is conservatively used for the factor on the Alloy 600 CGR in air. This is because Alloy 182 had the higher median value and a comparable uncertainty when compared to either Alloy 82 data or to the combined Alloy 82/182 data. The value of 4.1 for the exponent on change in stress intensity factor is taken directly from equation (4) of NUREG/CR-6721 [13] for the Alloy 600 FCG rate in air.

The fatigue stress range, stress ratio, and cycles per year are taken from evaluations of vendor calculations. As discussed further in Section 4, Benchmarking, the effects of FCG on leak probability is fairly small as compared to PWSCC. Therefore, the sensitivity of the fatigue stress input values is also very small.

3.4 Weld Residual Stress

For large thick-walled piping welds, the maximum residual stress would not be expected to exceed about 40 ksi (as shown in Figure 5-2 of the pc-PRAISE Users Manual [11]). However, for the multiple repairs and nearly double-V weld configuration in the Virgil C. Summer outlet nozzle weld [8], the residual stress could be much higher. Finite-element analyses of Alloy 82/182 butt welds for the MRP [14] show that the Virgil C. Summer conditions could easily result in mean stress values of 70 ksi. Residual stress was taken to be uniform through the thickness of the pipe based on ASME values. The selected value for Virgil C. Summer (70 ksi) would also represent a realistic upper limit on the values previously used for vessel head penetrations (see Figure 4-1 of WCAP-14901 [2]).

The MRP sponsored finite-element analyses [14] also showed that residual stress would be significantly lower for more typical double-V weld configurations without weld repairs. These lower residual stress values are also consistent with those used previously in the pipe butt weld safety evaluation for Westinghouse and CE plant designs [9]. It should be noted that all the butt welds in PWRs in the United States are single-V welds.

3.5 Weld Geometry and Operating Conditions

Weld geometry (such as, diameter and thickness dimensions) and operating conditions (such as, temperature and pressure) are taken from vendor calculations. This information, as well as deadweight and thermal stress and pressure stress, has been used previously in the evaluation in the leak in the reactor pressure vessel outlet nozzle weld at Virgil C. Summer [8]. Similar information was reported in the deterministic pipe butt weld safety assessment for Westinghouse and CE plant designs [9], and B&W designs [10].

3.6 Critical Flaw Size Limits

To determine when the butt weld would fail, the failure modes from the PFM models for piping RI-ISI [3] are used directly. The SIF calculations for independent crack growth in both the depth and length directions are for uniform stress through the wall thickness in NRC's probabilistic

analysis code pc-PRAISE [11] for piping welds. While Section 2.3.3 of the pc-PRAISE Users Manual [11] states that the SIF correlations apply equally well to axially and circumferentially oriented cracks, the ductile failure criteria are different for through-wall axial versus circumferential flaws. The PFM models for RI-ISI that were used [3] also include options to determine failure based on the following:

1. Exceeding a user-specified crack length for any size leak

2. Flaw exceeding a user-specified weld length

3. Full break (ductile rupture) at user-specified load and frequency conditions

4. Credit for detection of a user-specified value (such as, LBB)

For calculation of small-leak probabilities (such as, flaw growth through the pipe wall thickness), the flaw depth limit is set to 99 percent of the mean thickness of the weld. For large-leak probabilities (based on quantified leak input), the circumferential flaw length limit is taken from an analysis of leak rate with flaw length. An example of this, based on various weld dimensions and operating conditions in Westinghouse and CE plant designs appears in Chapter 6 of EPRI Report TP-1001491 [9].

3.7 Effects of In-Service Inspection

The beneficial effects of ISI are modeled in the same way as in the PFM models for piping RI-ISI [3] and as in the NRC's pc-PRAISE Code [11] for piping welds. Specifically, only the flaws remaining after an ISI exam are left to cause failures later in life because flaws that are detected are assumed to be removed. A 50-percent probability of detecting a flaw 25-percent through the wall thickness forms the basis for detectable inspection and elimination of the flaw [2,3]. Existing flaws that would not have reached this detection limit remain and would be subject to growth and detection at the specified inspection interval. This typically is considered as a 10-year interval based on ASME Section XI specifications. The effects of subsequent inspections can be either cumulative or independent, which usually depends on the interval between inspections. Both approaches were investigated and showed no statistical difference in the final result at end of life. The input to these PFM models is selected to represent the inspection accuracy and frequency for a UT inspection performed in accordance with the requirements of Section XI of the ASME Code.

3.8 Axial Versus Circumferential Flaws

The ratio of the number of axial flaws to circumferential flaws was developed from actual service experience. There is actually a rather large data base of cracking that has been found in Alloy 600 and Alloy 82/182 materials over the past 20 years. The largest number of cases occurred in the CE small-diameter pipes, but there are also the recent experiences with the reactor vessel outlet nozzle regions. In all, there are over 100 cracking incidents, and only one or two circumferential flaws.

The PWSCC first occurred in a nozzle application in a CE plant in 1986 when a pressurizer steam space instrumentation nozzle at San Onofre Unit 3 developed a leak after about 10,000 hours of service. Failure analysis of the removed nozzle confirmed the leak resulted from intergranular stress corrosion cracking with no identified contaminants. The analysis concluded that high carbon content (0.07%), high strength (yield strength of 60.9 ksi), and residual stresses from the fabrication process resulted in low resistance to PWSCC. Two additional steam space nozzles were cracked but not leaking at the first refueling outage at San Onofre Unit 3 and two nozzles of the same heat cracked at St Lucie Unit 2 but were not leaking at the time of replacement (after 32,500 hours of service). Table 3-2 summarizes the complete history of PWSCC in Alloy 600 nozzles in CE plants.

Prior to 2003, all cracks in small-diameter nozzles and heater sleeves installed with partial penetration welds, with one exception, were axially oriented. Therefore, they were considered to be stable and not safety significant. In this one exception, a heater sleeve which had an axially oriented through-wall flaw also had part through-wall circumferential cracks. These cracks were not associated with the partial penetration weld, but were located outside of the pressurizer shell at a location where the sleeve had been deformed during post-installation reaming during the fabrication process. The sleeve should have been, but was not, replaced. Cracking occurred in this deformed area.

Cracking in Alloys 182 and 82 was not observed in operating PWR plants until the year 2000, when several incidents occurred. The first was in the outlet nozzle to pipe safe end weld of Ringhals Unit 4, in July of 2000. Several small axial cracks were found, and removed with a boat sample by EDM. The first cracking event actually was found in June of that year, in Ringhals Unit 3, but the indications were thought to be shallow artifacts, and the plant was allowed to remain in service without repair. In both cases, the welds were Alloy 182, and the cracks were axial.

The next major incident occurred in October 2000, when the Virgil C. Summer plant was found to have a through-wall flaw in the same region as the Ringhals plants–the reactor vessel outlet nozzle to pipe safe end weld. Ultrasonic tests performed on the pipe from the inside surface initially revealed a single axial flaw near the top of the pipe. Followup exams conducted in spring of 2002 revealed that there were several flaws, all but one of which were axial, and that the largest axial flaw was through-wall. The flawed region was removed, and a new spool piece welded in place, restoring this region to its original condition. Destructive examination of the spool piece removed showed that there were six additional axial flaws, and one shallow circumferential flaw.

In November of 2003, a surface indication was reported in the Alloy 182 weld region of the hot leg nozzle to surge line safe end weld at Three Mile Island. The UT inspection technique was part of a periodic Section XI inspection program. The indication was axially oriented, and confined entirely to the weld, which had been repaired during construction and not stress relieved. The depth of the indication was reported as approximately 45% of the wall thickness.

Therefore, in the large-diameter pipes, there have been a total of four pipes with cracks, with one circumferential flaw and four axial flaws. If these are added to the cracks found in small bore

pipes, the ratio of axial to circumferential flaws is about 100 to 1, so this ratio was used in the analysis.

Table 3-2
Service Experience with CE Small Bore Piping

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4
PFM UNCERTAINTIES AND BENCHMARKING

Specific items pertinent to the evaluation discussed in this report are highlighted and summarized in this section. As shown in Table 4-1, the uncorrelated uncertainties in the PFM models for butt weld leaks are consistent with those used for previous models [2,3]. The uncertainties along with the reference for NRC review and acceptance of the PFM models are:

- Uncertainties in parameters for PWSCC time to initiation and CGR from the models for Alloy 600 head penetration nozzles [2,4,5]
- Uncertainties in weld geometry and failure parameters from the models for piping RI-ISI [3,6])

The 1-sigma uncertainty factor for a log-normal distribution of the coefficient is calculated from Figure 4-1, which is Attachment 5 in the MRP PWSCC Crack Growth Expert Panel Meeting Minutes [12]. As indicated in Section 3.3, data analysis is used to develop the uncertainty for the weld FCG factor. The data is the ratio of FCG of the weld in a PWR environment to the FCG of Alloy 600 base metal in air. Specifically, 21 ratio values for Alloys 182 and 32 values for Alloy 82 in NUREG/CR-6721 [13] were analyzed to determine the log-normal uncertainty on FCG.

The benchmarking process for the butt weld PFM models includes checking the time to initiation and through-wall crack growth (such as, small-leak) failure probabilities calculated for the Virgil C. Summer plant at the time the vessel outlet nozzle weld leak was discovered (17 years) [8]. For the mean residual stress level as discussed in Section 3.3 and the PWSCC growth rates for Alloy 82/182 from the Expert Panel [12], the only input parameter available for this benchmarking process is the combined factor on time to initiation (see Section 3.1). A combined factor of 1.16 on PWSCC initiation time gives a calculated small-leak probability of about 37% for an axial through wall flaw after 17 years.

This probability is considered to be representative of the Virgil C. Summer leak experience because it is consistent with the calculated probabilities for vessel head penetration nozzle experience with large axial flaws. As shown in Table 4-2 in WCAP-14901 [2], the PFM calculated probabilities of the observed head penetration nozzle cracks at Donald C. Cook Unit 2 and at Ringhals Unit 2 were similar in magnitude. An added check of the PFM model, including revised inputs, for butt weld axial flaws was used to calculate a similar probability for the observed crack in the surge line to hot leg nozzle butt weld at the Three Mile Island plant. The results were based on a residual stress input of 70 ksi and correlated with observed plant information.

Table 4-1
Summary of Uncertainties Used for PFM Analysis of Alloy 82/182 Butt Welds

Input Information With Uncertainty	Distribution	Uncertainty Data	Ref. No.
Butt Weld Outside Diameter (inch)	Normal	SRRA for Pipe RI-ISI	3
Butt Weld Wall Thickness (inch)	Normal	SRRA for Pipe RI-ISI	3
Butt Weld Operating Temperature (°F)	Uniform	Alloy 600 Head Pen.	2
Butt Weld Max. Residual Stress (ksi)	Normal	Alloy 600 Head Pen.	2
Butt Weld Pressure Hoop Stress (ksi)	Normal	SRRA for Pipe RI-ISI	3
Deadweight and Thermal Stress (ksi)	Log-Normal	SRRA for Pipe RI-ISI	3
Hoop/Axial Fatigue Stress Range (ksi)	Log-Normal	Vendor Calculations	9 and 10
Hours at Temperature per Year	Normal	Alloy 600 Head Pen.	2
PWSCC Initiation Time Coefficient (hrs)	Log-Normal	Alloy 600 Head Pen.	2
Initiation Activation Energy (cal/mole)	Normal	Alloy 600 Head Pen.	2
Crack Depth at Initiation (inch)	Normal	Alloy 600 Head Pen.	2
Crack Length at Initiation (inch)	Normal	Alloy 600 Head Pen.	2
PWSCC Growth Rate Coefficient (in/hr)	Log-Normal	MRP Expert Panel Data	12
PWSCC Rate Activation Energy (cal/mole)	Normal	Alloy 600 Head Pen.	2
Weld FCG Factor on Alloy 600 in Air	Log-Normal	Analysis of Data in ANL Report	13
Flow Stress for Full Weld Break (ksi)	Normal	SRRA for Pipe RI-ISI	3
Hoop/Axial Stress Evaluation Condition (ksi)	Log-Normal	SRRA for Pipe RI-ISI	3

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Figure 4-1
Uncertainty in PWSCC Crack Growth Curves from Alloy 600 MRP Expert Panel [12]

A combined factor of 7.06 on PWSCC initiation time gives a calculated small-leak probability of about 0.4% for a circumferential flaw after 17 years of operation at Virgil C. Summer [8]. This probability is consistent with the observation that only about one percent of the observed flaws in Alloy 82/182 weld metal are circumferentially oriented [15]. Figure 4-2 shows the calculated small-leak probabilities for both the axial and circumferential flaws at Virgil C. Summer. This figure also shows a somewhat smaller probability for a large circumferential leak of 800 gpm, which is the average of the leak rates from 100 gpm to 1500 gpm that are typically used for evaluating a medium-break loss-of-coolant accident (LOCA) [16]. This calculated large-leak probability is also consistent with the expectation that large-leak probabilities should be somewhat less than small-leak probabilities.

To verify the PWSCC and FCG calculations in circumferential flaws after flaw initiation, comparable PFM calculations were performed with the butt weld PFM models and the previously verified PFM models for piping RI-ISI [3]. To get FCG of any measurable effect relative to PWSCC-induced crack growth, the PWSCC loading had to be significantly reduced and the fatigue loading and cycles significantly increased.

The 40-year probabilities calculated both without and with the effects of ISI every 10 years are compared in Table 4-2. For small leak with fatigue, small leak without fatigue, and large leak with fatigue, the difference in the calculated probability values at 40 years is less than the standard deviation. This standard deviation reflects the uncertainty in the Monte-Carlo simulation for a small number of trials (typically less than 1000).

The results of Table 4-2 illustrate that the leak probabilities that were calculated for the evaluations summarized in this report, which include the revised PWSCC and FCG data [12], when compared with the leak probabilities independently calculated with the verified PFM models for piping-weld RI-ISI [3] are statistically identical. This conclusion of statistically insignificant differences is based on the standard deviation as calculated by the NRC-approved PFM code [3]. The conclusion of the benchmark is that the PFM models used to calculate the leak probabilities for axial and circumferential flaws in Alloy 82/182 butt welds in piping nozzles have been benchmarked with existing failure (small-leak) data and independently verified.

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Figure 4-2
Calculated Leak Probabilities After Benchmarking With Virgil C. Summer Reactor Pressure Vessel Outlet Nozzle Weld [8] Without Inspection

Table 4-2
Independent Verification of Butt Weld Circumferential Flaw Probabilities

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5 PFM CASES CONSIDERED AND RESULTS

5.1 Cases Considered

Probabilistic fracture mechanics evaluations were performed to address the identified degradation mechanisms of PWSCC and FCG on dissimilar metal butt welds. The analyses performed considered the limiting butt welds in large diameter pipes and smaller diameter pipes based on the deterministic evaluations for the Westinghouse [9], CE [9], and B&W [10] NSSS designs. Sensitivity studies were performed for key input parameters including residual stress due to fabrication history, and the effects of ISI (accuracy and frequency). The input value for residual stress for the overall assessment was based on no weld repairs in contrast to the evidence available to support the value of 70 ksi used in the benchmark studies. The locations selected for assessment were based on the dominating locations identified in the deterministic evaluations. Additional locations, such as the cold leg nozzles and pump welds, were of lesser concern based on the deterministic assessment but remain areas to consider for follow-on activities. The assessment of the locations addressed, therefore, are considered to be bounding of the remaining locations identified in the deterministic assessment provided there is not available evidence indicating weld repairs were made. Results of the PFM evaluations of the limiting butt welds are summarized in this section.

The PFM results are applicable to PWR plants that were designed by B&W, CE, and Westinghouse and contain Alloy 82/182 safe end welds. The evaluation summarized in this report considers Alloy 82/182 as weld metal, it does not apply to base metal. The following specific locations, including identification of NSSS design agents, were considered:

- Decay heat nozzle (B&W)
- Reactor vessel outlet nozzle to safe end weld (Westinghouse)
- Surge line hot leg nozzle to safe end weld (CE)
- Pressurizer surge nozzle to safe end weld (B&W, CE, Westinghouse)
- Pressurizer spray nozzle to safe end weld (CE, Westinghouse)
- Pressurizer safety and relief nozzle to safe end weld (CE, Westinghouse)
- Steam generator inlet nozzle to safe end weld (CE)
- Shutdown cooling nozzle to safe end weld (CE)

The PFM evaluations utilized the methodology, computer tools, and parameter uncertainties discussed in Sections 3 and 4 of this report. Evaluations for each location considered the following failure modes:

- Small axial leak
- Small circumferential leak

In order to develop risk calculation, it is necessary to determine the probability of an event, as well as the conditional core damage frequency as a result of that event. The evaluations presented herein were aimed at consideration of the first occurrence of a leak, as well as its progression to a large leak.

This section provides the results of the probabilistic calculations, which are linked with conditional core damage frequency values in Section 6. It is important to mention that the large axial leak is not considered, because the axial flaw will not propagate beyond the width of the weld, since the material at either end is not susceptible to PWSCC. For the purposes of this evaluation, the width of the weld is limited to the thickness of the pipe.

5.2 Results

Results of the PFM evaluation are provided in Table 5-1 and Figures 5-1 through 5-9.

Table 5-1
Summary of 40-Year Leak Probabilities

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The probabilistic cases were chosen from the most limiting results of the deterministic calculations [9,10], that is, those locations which had the shortest time between a 1 gpm leak, and a failure of the pipe. The calculations were carried out over a period of 60 years, as seen in the figures. Results were reported at both 40 and 60 years. As may be seen in each of the figures, results were obtained for a number of plants, again taken from the most limiting plants from the deterministic results. Of the results, only the highest ranked plant was reported in Table 5-1. It is also important to mention that Table 5-1 does not include the benchmark plant results for the reactor vessel outlet nozzle, since mitigating actions were taken.

As seen in Table 5-1, the locations which have the highest probability of leakage from an axial flaw after 40 years of service are the Westinghouse reactor vessel outlet nozzle and the spray nozzle for a CE-designed unit. For a circumferential flaw, the highest probabilities are for the Westinghouse reactor vessel outlet nozzle, and the pressurizer surge nozzle for the B&W design. Note that the probabilities of leakage are much higher for axial flaws than for circumferential flaws, which has been borne out by service experience.

Figures 5-1 through 5-8 provide plots of leakage probability versus time for each of the butt weld regions evaluated. Each figure provides the results for axial flaws in the first figure and circumferential flaws in the second figure. Results for the four or five most limiting units (from the deterministic work) as are each provided as separate curves. Results are slightly different for the various units, primarily due to differences in geometry and loadings for the various plants.

It is important to note that the probability of leakage for circumferential flaws is much lower that for axial flaws, even though at first glance the figures seem to be similar. The scale has been expanded for the circumferential flaws so the plant differences are legible.

The results for the B&W designs are included in the results shown in Figures 5-9 through 5-11. These analyses were carried out by AREVA personnel for proprietary reasons and also used the Westinghouse PFM tool discussed in Sections 3 and 4. These cases were completed, for the governing locations that resulted from the deterministic work. Only axial flaws were considered, and the results are similar to those for the other designs.

5.3 Sensitivity Studies

The sensitivity results represent a PFM evaluation on a nozzle weld that was considered to be representative of the effects that would be observed on the other nozzle welds considered in this study. The residual stress input in the sensitivity study considers the effects of weld repairs and is based on the maximum residual stress values developed by the ASME Section XI Task Group for Piping Flaw Evaluation.

The results of the sensitivity study for ISI accuracy and frequency are shown in Table 5-2. Since the effect of inspection frequency and accuracy is fairly small, the 10-year ISI with standard accuracy was selected for the remaining analyses. The results of the sensitivity study for residual stress in Table 5-3 show a very significant effect. A mean residual stress of 20 ksi was chosen to be conservative but not unrealistic.

An additional check of the PFM methodology was performed for the Three Mile Island (B&W) hot leg surge nozzle weld in Figure 5-9 and showed good correlation to the observed 45% through-wall flaw found at 24 years of life. A failure probability of 23% was obtained from the PFM analysis. The TMI hot leg nozzle was repaired and results of the benchmark would not necessarily be representative of the remainder of the B&W fleet.

Table 5-2
In-Service Inspection Sensitivity Study

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Comparing the leak probability of an axial flaw versus a circumferential flaw for all cases in Table 5-3 shows that the axial flaw leak is much more likely to occur. However, the more probable leaks from axial flaws would be very small because the length of the flaw is limited by the length of the weld. Only the leaks from circumferential flaws, which could grow around the circumference, are large enough to be a safety concern. As shown in Figure 6-1 of the deterministic safety assessment of Westinghouse nozzle welds [9], the maximum axial-flaw leak rate in the large reactor vessel outlet nozzle weld would be less than 2 gpm.

Table 5-3
Residual Stress Sensitivity Study

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Figure 5-1
RPVON Axial Small Leak Without Benchmark

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Figure 5-2
RPVON Circumferential Small Leak Without Benchmark

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Figure 5-3
Safety and Relief Axial Small Leak

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Figure 5-4
Safety and Relief Circumferential Small Leak

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Figure 5-5
Spray Axial Small Leak

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Figure 5-6
Spray Circumferential Small Leak

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Figure 5-7
Surge Pressurizer Nozzle Axial Small Leak

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Figure 5-8
Surge Pressurizer Nozzle Circumferential Small Leak

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Figure 5-9
B&W Three Mile Island Surge Hot Leg

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Figure 5-10
Comparison of Maximum Axial Small Leak Probabilities in Various Nozzle Welds

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Figure 5-11
Comparison of Maximum Circumferential Small Leak Probabilities in Various Nozzle Welds

6
RISK EVALUATION

The NRC-developed definition for risk-informed evaluations is summarized as utilization of "insights derived from probabilistic risk assessments used in combination with deterministic system and engineering analysis to focus licensee and regulatory attention on issues commensurate with their importance to safety." Guidelines for determining the significance of changes in risk as a result of identified degradation mechanisms, such as PWSCC, are provided in U.S. NRC RG 1.174 [1]. Four basic steps are identified in the guidelines:

1. Define Proposed Change

2. Perform Engineering Analysis

3. Define Implementation and Monitoring Program

4. Submit Proposed Change

Of the four steps only one represents a change in the basis for the plant design, Step 2, "Performing the Engineering Analysis." This is the focus for the evaluation summarized in this report. This element includes performing an evaluation to show that the fundamental safety principles on which the plant design was based are not compromised and that sufficient safety margins are maintained. The engineering analysis includes both traditional deterministic analysis and PRA. The deterministic assessment [9,10] identified critical locations that were likely to cause safety impacts to the plant. Probabilistic fracture mechanics techniques as discussed in previous sections were used to determine more realistic values at these locations for postulating the growth of a flaw through the wall. This through-wall flaw was equated to the safety-related consequences (such as, PRA) that would result from either a small loss-of-coolant accident (SLOCA) or medium LOCA (MLOCA). The quantitative assessment of risk and "insignificant change" in risk based on core damage frequency (CDF) and large early release frequency (LERF) are summarized as follows:

- Total plant CDF is less than 10^{-5} per reactor year
- Total plant LERF is less than 10^{-6} per reactor year
- Change in CDF is less than 10^{-6} per reactor year
- Change in LERF is less than 10^{-7} per reactor year

The CDF was calculated by:

$$CDF = IE * CCDP$$

where:

CDF = Core damage frequency from a failure (events per year)

CCDP = Conditional core damage probability

IE = Initiating event frequency (in events per year) = pipe failure frequency per year

A similar calculation could be preformed for determining LERF by substituting appropriate values for the CCDP term.

The CCDP for SLOCA and MLOCA were represented by a generic value of 0.003. This value is a reasonably conservative screening value for this application. The overall risk value combines the conditional consequences from failure of the mitigation flow path and the leak probabilities converted to initiating event frequencies from the PFM. This is done to determine the potential change in risk, in terms of the mean value of CDF based on point estimates. Effects of uncertainties were considered in the values of both the leak probabilities and the conditional consequences. The uncertainty in the CCDP was set at an order of magnitude range within the 5% and 95% confidence bounds on a log-normal distribution about the identified mean value of 0.003.

The PRAs typically determined the impact (such as, CDF/LERF) for plant conditions represented by disabling leaks, SLOCAs, MLOCAs, large LOCA (LLOCA), or pipe rupture. For this study, leak rates to represent these conditions are equated to a calculated value for system disabling, less than 100 gpm SLOCA, 100 to 1500 gpm for MLOCA, and greater than 1500 gpm for LLOCA [16]. The makeup capacity of a plant typically mitigates system disabling leaks and the plant is brought to a safe shutdown condition through existing operating procedures (such as, normal plant shutdown). For this evaluation, the mean CCDP of 0.003 was intended as a bounding assessment.

The values for total plant change in risk were determined by summing the increase in risk from the individual nozzle contributions. A summary of 40- and 60-year risk increase by individual weld nozzle location is provided in Table 6-1.

Table 6-1
Summary of 40- and 60- Year Risk Increase by Nozzle

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The total increase in risk based on the critical contributing nozzle locations identified due to butt weld PWSCC for each NSSS design is included in Table 6-2. It accounts for the number of nozzles per plant design. The increase in risk from other Alloy 82/182 butt weld nozzles is not considered based on the Alloy 82/182 results of the deterministic assessments [9,10] and conservatism of the bounding PRA value. The risk values calculated for CDF were compared with the guidelines for an "insignificant change" in risk given in RG 1.174. The results of the evaluation of the point estimate risk with inspection are summarized in Tables 6-1 and 6-2. These results demonstrate that the RG 1.174 criterion is met for all cases for CDF. If the 95% upper confidence bound in the CCDP of $\sqrt{10}$ is applied, the total risk results would still be insignificant per the RG 1.174 guidelines. It must be noted that this conclusion is driven by a calculated value for leak probability at 40 years that did not consider any mitigative action, such as detection and physical modifications, to correct or eliminate the likelihood of the leak occurring. Relative risk results are also summarized for the inspection sensitivity study in Table 6-3. The relative change in risk due to increasing frequency of inspections to yearly and improving the quality of inspection by a factor of 2.5 results in a factor of 4 decrease in leak probability. This is an insignificant difference. The conclusion of this study is that the change in relative risk with inspection would be relatively insignificant compared to the uncertainties in probability and consequence.

Table 6-2
Plant 40- and 60-Year Total Increase in Risk Summary by Nozzle and NSSS Design (events per year)

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Table 6-3
Inspection Sensitivity Study

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A similar conclusion would be reached for LERF. The LERF values for an individual flow path are typically less than those for CDF. If one conservatively assumes the value for LERF is identical to the value used in this assessment for CDF and then compares it to the regulatory guidelines, the conclusion would be that the result is acceptable when compared to the guideline for "insignificant change" in LERF given in RG 1.174 for the CE, Westinghouse, and B&W nozzles.

The results show that the small axial leak has a much higher leak probability than the small circumferential leak, but the consequences are much less significant. The typical makeup capacity of the plant would address the potential maximum 2 gpm leak from an axial flow, as predicted by the deterministic evaluation [9]. Based on the PFM results, plant leak detection would have sufficient time to detect the leak and initiate mitigative actions to prevent a leak from progressing to a large break such as would cause an LLOCA. It is, therefore, concluded that the risk calculated for the SLOCA and MLOCA would comprise the total contributing increase in risk when comparing to RG 1.174 guidelines for this assessment.

The uncertainty in calculated probabilities of leak from circumferential flaws could have a significant effect on the risk increase due to PWSCC in the bi-metallic welds. The largest contributor to this uncertainty is the 1% ratio of circumferential flaws to axial flaws that was used in the benchmarking calculations in Section 4. Some of this uncertainty has been compensated for by using the higher probability of a small leak (through-wall circumferential flaw) for the probabilities of the larger leaks of concern. Figure 4-2 shows these differences in calculated leak probabilities for the Virgil C. Summer vessel outlet nozzle weld that was used for benchmarking.

There is little data on observed circumferential flaws. Therefore, a statistical upper confidence bound on the percentage should be no more than a factor of 6 higher than the 1% that was used. Even with this upper bound factor of 6 increase in probability of large leaks in circumferential flaws, the total risk increase due to PWSCC in all the bimetallic butt welds would still be classified as insignificant per the guidelines of RG 1.174 [1].

The total plant CDF and LERF are not specifically calculated since the specific plant values are not part of the evaluation. A review of plant-specific values would likely show acceptance for 40 years for all Westinghouse, CE, and B&W nozzles. The value for 60 years would need plant-specific evaluations to draw concise conclusions but it is likely that the plant specific values would be less than those summarized in Table 6-2.

This study demonstrates that the relative change in risk based on bounding conditions for the locations likely to yield a risk increase is within RG 1.174 guidelines.

7 SUMMARY OF RESULTS, CONCLUSIONS, AND RECOMMENDATIONS

The evaluations documented in this report have been intended to cover all the Alloy 82/182 butt weld locations in operating PWRs in the USA. Two companion reports [9,10] have been prepared documenting deterministic treatments of these welds. These deterministic results, as well as complementary work to provide input on the effects of repairs [4,17,18] and crack growth modeling [12], have been brought together in this report. The three goals of this work were the following:

- Quantify the probability of leakage from both axial and circumferential flaws
- Assess the impact of the calculated change in core damage risk per the RG 1.174 guidelines [1]
- Develop a recommendation as to the adequacy of the current ASME Section XI inspection requirements for these regions

The methodology used for the evaluations summarized in this report is consistent with that used in previous submittals that have been reviewed and approved by the NRC. Examples are the Westinghouse Owners Group treatment of the probability of head penetration cracking [2] and the Westinghouse risk-informed inspection approach [3]. Key effects treated in this work include crack initiation, crack growth, conditions for failure of the pipe, and the consequences of a range of piping leaks on the conditional CDF. Note that cracking in only the weld metal was considered in this study. There are a few cases where the safe end material is Alloy 600, and these were not treated by this evaluation.

Since the existing PFM models for reactor vessel head penetrations [2,4,5] already included crack growth due to PWSCC, the new correlation for weld metal (Alloy 82, 132, and 182) developed by the MRP PWSCC Crack Growth Expert Panel [12] was used directly. Only the input parameters for the original Alloy 600 model needed to be updated. The existing PFM models for FCG for piping RI-ISI [3,6] are used directly for the PFM models for butt welds. Figures 14(a) and 14(b) of the ANL Report [13] indicate that FCG of Alloy 82/182, respectively, is best characterized as a factor on the Alloy 600 CGR in air. Analyses of this data are used to develop the mean value and uncertainty for the factor on weld FCG. The benchmarking process to address these enhancements is discussed in full in Section 4 of this report. The conclusion is that the PFM models used to calculate the leak probabilities for axial and circumferential flaws in Alloy 82/182 butt welds in piping nozzles have been benchmarked with existing failure (small-leak) data and independently verified to produce accurate results.

The leak probability calculated for axial flaws, using PFM methodologies with a 10-year inspection, ranges from 5.00E-05 to 1.69E-02 at 40 years of plant life. The leak probability for a circumferential flaw with a 10-year inspection ranges from 2.70E-08 to 2.00E-04 at 40 years of plant life. Probabilities for larger leaks that would correspond to small, medium, or large LOCAs would be smaller.

A comparison of axial versus circumferential leak probabilities shows that the axial probability is consistently higher than the circumferential probability, by more than two orders of magnitude. The axial flaw length would be limited in extent to the interface with material not susceptible to PWSCC, which is consistent with service experience. This evaluation considered this length to be equal to the thickness of the weld, which is a conservative assumption.

The results of the assessment showed that the change in total plant risk was well within the RG 1.174 guidelines for "insignificant change" when considering either CDF or LERF for a 40-year plant life. Calculated CDF point estimate values for a 40-year life ranged from 1.85E-08 to 8.74E-08. These values for total plant change in risk were determined by combining the worst-case leak probability for each location with a generic CCDP value of 3.00E-03 and combining the contributions from the individual nozzles. The calculation for a plant-specific application would be lower. The consequences for LERF would be lower than those for core damage, based on actual plant data. Therefore, the same value of 3.0E-03 was used as a bounding case and the change in risk for LERF would also be insignificant.

The existing analyses provide a baseline from which to draw additional risk assessment insight. As the existing analyses contain a limited set of sensitivity studies for these parameters that were modified as a result of emerging industry studies from the approved base model, additional analyses may be needed to provide a better understanding of the impact of uncertainty. It is anticipated that these studies will be completed as part of the on-going industry effort.

There are several items that would contribute to the conservatisms in the leak probabilities calculated. Two that result in the greatest overall contribution are the treatment of weld residual stress and credit for leak detection. Residual stresses are represented as the peak through-wall value conservatively applied over the entire thickness. The capability for leak detection at all plants is required to be at least 1 gpm. However, the actual leak detection capability is now significantly better than this value, typically by an order of magnitude. The results presented in this report do not consider this factor in determining the leak probability of either axial or circumferential flaws for any potential break size or consequence. In addition, no consideration of plant mitigative action in addressing leaks was included in the determination of the conditional core damage probabilities.

Critical conclusions from the evaluation are:

- The fabrication history of the weld is a key contributor.
- Changes in inspection frequency or improvements in capability or accuracy have only a small benefit for the locations with the highest leak probabilities.
- Risk results do not justify any required changes in the current 10-year ASME Code Section XI inspection interval, as long as all Alloy 182/82 locations are included.

A review of the critical nozzle leak and risk assessment results would suggest that the reactor pressure vessel outlet nozzle is the most critical when considering potential leaks, plant reliability, and plant safety. The pressurizer surge line, pressurizer spray, decay heat, and pressurizer safety and relief nozzle would follow in order of concern.

8 REFERENCES

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4. *NRC Staff Acceptance of Generic Industry Response to Generic Letter 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Head Penetrations*, NEI letter dated March 30, 1999 (included as Appendix A).

5. *Review of Generic Response to the NRC Requests for Additional Information Regarding Generic Letter 97-01*, NRC letter dated March 21, 1999 (included as Appendix B).

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A
APPENDIX

NUCLEAR ENERGY INSTITUTE

David J. Modeen
DIRECTOR, ENGINEERING
NUCLEAR GENERATION DIVISION

March 30, 1999

TO: NEI Administrative Points of Contact

SUBJECT: NRC Staff Acceptance of Generic Industry Response to Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Head Penetrations*

By early summer 1999, the NRC staff intends to issue PWR licensees close-out letters to GL (Generic Letter) 97-01 if the industry integrated program was used as a basis for the licensee response to the NRC request for additional information. This conclusion was provided to NEI in the enclosed NRC letter. **Please provide this information to individuals in your organization responsible for addressing the generic letter.**

GL 97-01 was issued to PWR licensees in April 1997. Licensees subsequently provided GL responses using owners group and NEI guidance. All PWR licensees received a request for additional information (RAI) from the NRC staff. The Alloy 600 Issue Task Group of the PWR Materials Reliability Project, with Owners Group support, developed a generic response to the RAIs which was submitted to the NRC for review in December 1998. Appropriate portions of this generic RAI response were referenced by licensees in their plant specific RAI response.

The NRC letter to NEI states:

> "The initial generic responses to GL 97-01, when taken in context with the information in your response of December 11, 1999, provide an acceptable approach for addressing the potential for PWSCC to occur in the VHP nozzles of PWR facilities. The staff, therefore, concludes that the integrated program proposed by NEI for VHP nozzles is acceptable, and that the licensees responding to the GL may refer to the integrated program as a basis for assessing the postulated occurrence of PWSCC in the PWR-design VHP nozzles."

If you have questions, please contact Kurt Cozens at (202) 739-8085 or koc@nei.org.

Sincerely,

David J. Modeen

David J. Modeen

KOC/avw
Enclosure

c: EPRI Alloy 600 Issue Task Group

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B
APPENDIX

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 21, 1999

Mr. David J. Modeen
Director of Engineering
Nuclear Generation Division
Nuclear Energy Institute
1776 I Street, NW
Washington, DC 20006-3708

SUBJECT: REVIEW OF GENERIC RESPONSE TO THE NRC REQUESTS FOR ADDITIONAL INFORMATION REGARDING GENERIC LETTER 97-01

Dear Mr. Modeen:

On April 1, 1997, the staff issued Generic Letter (GL) 97-01, "Degradation of CRDM/CEDM Nozzle and other Vessel Closure Head Penetrations," to the industry requesting that addressees provide a description of the plans to inspect the vessel head penetration (VHP) nozzles at their respective pressurized water reactor (PWR) designed plants. With respect to the issuance of the GL, the staff required the addressees to submit an initial response within 30 days of issuance informing the staff of the intent to comply with requested information and a follow-up response within 120 days of issuance containing the technical details to the staff's information requests.

The industry's responses to the GL included both generic responses from the respective Pressurized Water Reactor Owners Groups (PWROGs), and a plant-specific response from each PWR owned by an NRC licensed utility. These generic responses were coordinated by the Nuclear Energy Institute (NEI), and included submittal of Topical Report BAW-2301 on behalf of the Babcock & Wilcox Owners Group members, Topical Report CE NPSD-1085 on behalf of the Combustion Engineering Owners Group members, and Topical Reports WCAP-14901 and WCAP-14902, on behalf of Westinghouse Owners Group members. Each of these reports provided the probabilistic failure model that was used to assess and rank the relative susceptibility of a participating plant's VHP nozzles to undergo primary water stress corrosion cracking (PWSCC) over time. The staff's review included a review of both the generic responses and plant-specific responses to the GL. During the review, the staff determined that some additional information was needed for completion of the review. The staff, therefore, issued a series of requests for additional information (RAIs) to each of the addressees covered under the scope of the GL. The RAIs included the following generic requests:

1. the probabilistic susceptibility ranking for a plant's VHP nozzles to undergo PWSCC relative to the rankings for the rest of the industry
2. a description of how the respective susceptibility models were benchmarked
3. a description of how the variability in the product forms, material specifications, and heat treatments used to fabricate a plant's VHP nozzles were addressed in the susceptibility models

4. a description of how the models would be refined in the future to include plant-specific inspection results

However, as was the case for the earlier responses to the GL, the staff recommended that the addressees combine their efforts with those of the NEI, the PWROGs, and the Electric Power Research Institute in order to prompt a coordinated, generic response to the RAIs.

The staff has received and completed its review of NEI's submittal of December 11. 1998, which provided the generic response to the RAIs on behalf of the PWR industry, and in addition proposed an integrated program on behalf of the industry. Your submittal accomplished the following objectives:

1. Clarified that only two susceptibility models are currently being adopted by the industry for the assessment of VHP nozzles in the PWR industry.
2. Confirmed that participating members of the CEOG had opted to use the susceptibility model designed by the Dominion Engineering Company as the basis for ranking the VHP nozzles in CE designed plants.
3. Amended the probabilistic ranking histograms to identify the plants which fell into the respective histogram ranking groups.
4. Addressed how each of the susceptibility models were benchmarked.
5. Addressed how each of the susceptibility models would be refined to include plant-specific inspection results.
6. Addressed how the variability in product forms, material specifications, and heat treatments used for fabrication of VHP nozzles were addressed in each of the respective susceptibility models.

The initial generic responses to GL 97-01, when taken in context with the information in your response of December 11, 1999, provide an acceptable approach for addressing the potential for PWSCC to occur in the VHP nozzles of PWR facilities. The staff, therefore, concludes that the integrated program proposed by NEI for VHP nozzles is acceptable, and that the licensees responding to the GL may refer to the integrated program as a basis for assessing the postulated occurrence of PWSCC in PWR-design VHP nozzles. It is anticipated that closure of the staff's reviews of the plant-specific responses will be completed in the near-term, and that a NUREG will follow which summarizes the staff's and industry's efforts in addressing this issue.

Sincerely,

Jack R. Strosnider, Director
Divisiof Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

C
APPENDIX

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 15, 1998

Mr. Lou Liberatori, Chairman
Westinghouse Owners Group Steering Committee
Indian Point Unit 2
Broadway & Bleakley Ave.
Buchanan, NY 10511

SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT WCAP-14572, REVISION 1, "WESTINGHOUSE OWNERS GROUP APPLICATION OF RISK-INFORMED METHODS TO PIPING INSERVICE INSPECTION TOPICAL REPORT"

The NRC staff has completed its review of the subject topical report which was submitted by the Westinghouse Owners Group (WOG) through the Nuclear Energy Institute (NEI) by letter dated October 10, 1997. The staff has found that this report is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the report.

Current inspection requirements for commercial nuclear power plants are contained in the 1989 edition of Section XI, Division 1 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), entitled *Rules for Inservice Inspection of Nuclear Power Plant Components.* WCAP-14572, Revision 1, provides technical guidance on an alternative for selecting and categorizing piping components into high safety-significant (HSS) and low safety-significant (LSS) groups for the purpose of developing a risk-informed inservice inspection (ISI) program as an alternative to the ASME BPVC Section XI ISI requirements for piping. The RI-ISI programs can enhance overall safety by focusing inspections of piping at HSS locations and locations where failure mechanisms are likely to be present, and by improving the effectiveness of inspection of components by focusing on personnel qualifications, inspection for cause, and the use of the expert panel. The WCAP provides details required to incorporate risk-insights when identifying locations for inservice inspections of piping, in accordance with the general guidance provided in Regulatory Guide (RG)-1.174 and RG-1.178.

The staff will not repeat its review of the matters described in the WOG Topical Report WCAP-14572, Revision 1, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. In accordance with procedures established in NUREG-0390, the NRC requests that WOG publish accepted version of the submittal, within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract and an -A (designating accepted) following the report identification symbol.

RECEIVED
JAN 2 [illegible] 1999
WOG PROJECT OFFICE

If the NRC's criteria or regulations change so that its conclusion that the submittal is acceptable are invalidated, WOG and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

Should you have any questions or wish further clarification, please call me at (301) 415-1282 or Syed Ali at (301) 415-2776.

Sincerely,

Thomas H. Essig

Thomas H. Essig, Acting Chief
Generic Issues and Environmental Branch
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Safety Evaluation

cc w/enc: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

WCAP-14572, REVISION 1, "WESTINGHOUSE OWNERS GROUP

APPLICATION OF RISK-INFORMED METHODS

TO PIPING INSERVICE INSPECTION TOPICAL REPORT"

Enclosure

TABLE OF CONTENTS

ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CRGR	Committee to Review Generic Requirements
EC	Erosion Corrosion
EPRI	Electric Power Research Institute
FAC	Flow-assisted Corrosion
FSAR	Final Safety Analysis Report
IGSCC	Intergrannular Stress Corrosion Cracking
ISI	Inservice Inspection
LERF	Large Early Relief Frequency
MOV	Motor-operated Valves
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
POD	Probability of Detection
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
RAW	Risk Achievement Worth
RCS	Reactor Coolant System
RG	Regulatory Guide
RI-ISI	Risk-informed Inservice Inspection
RRW	Risk Reduction Worth
SER	Safety Evaluation Report
SRP	Standard Review Plan
SRRA	Structural Reliability and Risk Assessment
WOG	Westinghouse Owners Group

SAFETY EVALUATION REPORT RELATED TO "WESTINGHOUSE OWNERS GROUP APPLICATION OF RISK-INFORMED METHODS TO PIPING INSERVICE INSPECTION" (TOPICAL REPORT WCAP-14572, REVISION 1)

1.0 INTRODUCTION

On October 10, 1997, Nuclear Energy Institute (NEI), on behalf of Westinghouse Owners Group (WOG), submitted Revision 1 of Topical Report, WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection," (Ref. 1) for review and approval by the staff of the U. S. Nuclear Regulatory Commission (NRC). Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection, " (Ref. 2) was included as part of that submittal.

WCAP-14572, Revision 1, provides technical guidance on an alternative for selecting and categorizing piping components as high safety-significant (HSS) or low safety-significant (LSS) groups in order to develop a risk-informed inservice inspection (ISI) program as an alternative to the American Society of Mechanical Engineers (ASME) BPVC Section XI ISI requirements for piping. Current inspection requirements for commercial nuclear power plants are contained in the 1989 Edition of Section XI, Division 1 of the ASME Boiler and Pressure Vessel Code (BPVC), entitled "Rules for Inservice Inspection of Nuclear Power Plant Components", (the Code). The risk-informed inservice inspection (RI-ISI) programs enhance overall safety by focusing inspections of piping at HSS locations and locations where failure mechanisms are likely to be present, and by improving the effectiveness of inspection of components because the examination methods are based on the postulated failure mode and the configuration of the piping structural element. WCAP-14572 provides details required to incorporate risk-insights when identifying locations for inservice inspections of piping, in accordance with the general guidance provided in Regulatory Guide (RG)-1.174 (Ref. 3) and RG-1.178 (Ref. 4).

The WOG has asserted that the WCAP methodology for RI-ISI is a detailed implementation document for ASME Code Case N-577 (Ref. 5). However, the staff has not evaluated Code Case N-577 to determine its acceptability. Also, the staff has not evaluated WCAP-14572 to determine if it is an acceptable document to meet the intent of Code Case N-577.

In developing the methods described in WCAP-14572, Revision 1, the industry incorporated insights gained from two plants, Millstone Unit 3 and Surry Unit 1. The staff's review of WCAP-14572 incorporates information obtained through technical discussions at public meetings and through formal requests for additional information to address the issues related to the analytical methods, observance of the application of the methods to the Surry pilot plant, review of the Surry RI-ISI application, independent audit calculations, and peer reviews of selected technical issues.

2.0 SUMMARY OF THE PROPOSED APPROACH

The scope of the RI-ISI program includes changes in the current ASME XI piping ISI requirements with regard to the number of inspections, locations of inspections, and methods of

inspections. The scope of the RI-ISI program does not include changes in the current ASME XI piping ISI requirements with regard to the inspection intervals and periods, acceptance criteria for evaluation of flaws, expansion criteria for flaws discovered, inspection techniques and personnel qualification. It should also be noted that augmented examination program for degradation mechanisms such as intergrannular stress corrosion cracking (IGSCC) and erosion-corrosion (EC) would remain unaffected by the RI-ISI program.

Page 4 (Section 1.1) of WCAP-14572 states that "This report provides an alternative inspection location selection method for nondestructive examination (NDE) and does not affect current Owner-defined augmented programs." For RI-ISI programs whose scope incorporates augmented inspection programs, the effect of the current augmented programs on risk should be addressed. In most circumstances, the staff believes that the current augmented programs would be found acceptable. However, should the RI-ISI analysis identify that improvements to the augmented programs are warranted to maintain risk at acceptable levels, then those changes should be integrated into the respective programs.

The proposed approach is specifically for the NDE of Class 1 and 2 piping welds, but also includes Class 3 systems and non-Code class components found to be HSS in the risk evaluation. As stated by the Westinghouse Owners Group (WOG), other non-related portions of the Code will not be affected by implementation of WCAP-14572, Revision 1, approach.

The RI-ISI process includes the following steps:

- scope definition
- segment definition
- consequence evaluation
- failure probability estimation
- risk evaluation
- expert panel categorization
- element/NDE selection
- implementation, monitoring, and feedback

3.0 EVALUATION

For this safety evaluation, the NRC staff reviewed the WOG RI-ISI methodology, as defined by WCAP-14572, Revision 1, and its Supplement 1, with respect to the guidance contained in RG 1.178 and Standard Review Plan (SRP) Chapter 3.9.8 (Ref. 6) which describes the acceptable methodology, acceptance guidelines, and review process for proposed plant-specific, risk-informed changes to ISI programs for piping components. Further guidance is provided in RG 1.174 and SRP Chapter 19.0 (Ref. 7) which contains general guidance for using Probabilistic Risk Assessments in risk-informed decision-making.

3.1 Proposed Changes to the ISI Programs

Under the ASME Code, licensees are required to perform inservice inspection (ISI) of Category B-J and C-F piping welds, as well as Examination Category B-F dissimilar metal welds, during

successive 120-month (10-year) intervals. Currently, 25% of all Category B-J piping welds greater than 1-inch nominal diameter are selected for volumetric and/or surface examination on the basis of existing stress analyses. For Category C-F piping welds, 7.5% of non-exempt welds are selected for surface and/or volumetric examination. Under Examination Category B-F, all dissimilar metal welds require volumetric and/or surface examination.

Pursuant to Title 10, Section 50.55a(a)(3)(i), of the *Code of Federal Regulations* (10 CFR 50.55a(a)(3)(i)), licensees proposing to use WCAP-14572 methodology would propose an alternative to the ASME Code examination requirements for piping ISI at their plants. As stated in Section 1.2 of WCAP-14572, Revision 1, the RI-ISI program is intended to improve ISI effectiveness by focusing inspection resources on HSS locations where failure mechanisms are likely to occur. Therefore, the proposed approach meets the intent of ASME Section XI that the flaws are found before they lead to leakage and therefore the approach provides an acceptable level of safety.

Augmented examination program for degradation mechanisms such as IGSCC and EC would remain unaffected by the RI-ISI program. As stated in the WCAP-14572 (page 80, Section 3.5.5) and reiterated in the public meeting (item 11, Ref. 8) with Westinghouse on September 22, 1998, no changes to the augmented inspection programs are being made with the proposed change to the ASME Section XI Program. For calculating risk rankings, augmented programs such as erosion-corrosion and stress corrosion cracking programs are credited when the augmented program is deemed adequate to detect relevant degradation mechanisms. Augmented programs are also credited in the change of risk evaluation for both ASME Section XI programs and RI-ISI programs.

Sections 1.1 and 1.4 of WCAP-14572, Revision 1, describe the proposed changes to the ISI program that would result from applying this methodology. Details of the proposed changes (that is, the specific pipe systems, segments, and welds, as well as the specific revisions to inspection scope, locations, and techniques) are plant-specific and, therefore, are not directly applicable to this evaluation. Section 3.2 of WCAP-14572 describes the process for identifying the piping systems to be included in the scope of the RI-ISI program. Plant functions are considered in the expert panel review process during the consequence evaluation. In response to the staff open item 8(a) (Ref. 9), WCAP-14572 is being revised (Ref. 8) to state that the safety functions of the system and piping segment being reviewed should be presented to the expert panel to ensure that the expert panel specifically addresses the relationship between the systems and piping being evaluated and their associated plant safety functions. WCAP Sections 3.5.2 and 3.5.3 address how industry and plant-specific experience are considered as part of the evaluation process. Finally, Sections 4.4 and 4.5 of WCAP-14572 provide examples from the pilot studies of revisions to inspection scope, locations, and techniques.

3.2 Engineering Analysis

According to the guidelines in RGs 1.174 and 1.178, the licensees proposing an RI-ISI program should perform an analysis of the proposed changes using a combination of engineering analysis with supporting insights from a probabilistic risk assessment (PRA). For the RI-ISI program, engineering analysis includes determining the scope of piping systems included in the RI-ISI program, establishing the methodology for defining piping segments, evaluating the failure

potential of each segment, and determining the consequences of failure of piping segments. The following subsections discuss each of these aspects in greater detail.

3.2.1 Scope of Piping Systems

In accordance with the guidelines in Section 1.3 of RG 1.178, the staff has determined that full scope and partial scope options are acceptable for RI-ISI programs for piping. The full scope option includes ASME Class 1, 2, and 3 piping and piping whose failure would compromise safety related structures, systems, or components (SSC), and non-safety related piping that are relied upon to mitigate accidents or whose failure could prevent safety-related SSC to perform their function or whose failure could cause a reactor scram or actuation of a safety-related system. For the partial scope option, a licensee may elect its RI-ISI program for a subset of piping classes, for example, Class 1 piping only.

Section 3.2 of WCAP-14572, Revision 1, describes the scope of systems to be considered in an RI-ISI program. WCAP-14572 identifies three criteria for system selection. Criterion 1: all Class 1, 2, and 3 systems currently within the ASME Section XI program; Criterion 2: piping systems modeled in the PRA; and Criterion 3: balance of plant fluid systems determined to be of importance (mainly on the basis of NEI guidance for implementation of the Maintenance Rule with respect to safety significance categorization). The Maintenance Rule scope definition is used to provide a starting point for the determination of the scope of the RI-ISI program.

Section 2.3 of WCAP-14572 states that the scope incorporates piping segment cutsets that cumulatively account for about 90 percent of the core damage frequency attributed from piping alone.

In addressing the exclusion of piping systems from the scope of the RI-ISI program, Section 3.2 of WCAP-14572 includes the following explanation:

> "Twenty-one systems were selected to be evaluated in more detail for the representative WOG plant. The remaining systems are excluded from the scope of the risk-informed ISI program. These systems are not addressed by ASME Section XI, but some were considered by the PRA (such as emergency diesel jacket water, containment instrument air, and instrument air). However, each of these systems was reviewed by the plant expert panel using the same criteria as in the determination of risk-significance for the Maintenance Rule. In addition, the consequences postulated from the loss of any of these systems from a pipe failure were determined not to be significant. Therefore, these systems in their entirety, were determined to be outside the scope and not further evaluated."

In order to allow for partial scope, the next revision of WCAP-14572 will add the following statement in Section 3 and 3.2 as stated on page 264 of Ref. 8:

> "A full scope program is recommended because a greater portion of the plant risk from piping pressure boundary failures is addressed in the risk-informed ISI program versus current ASME Section XI requirements since the examination are now placed in several high-safety-significant piping segments that are not currently examined by the current Section XI approach. However, a partial scope evaluation may be performed given that the evaluation

includes a subset of piping classes, for example, ASME Class 1 piping only, including piping exempt from the current requirements."

The staff finds acceptable the discussion of scope since this definition is consistent with guidance provided in RG 1.178 and SRP Chapter 3.9.8. However, the staff notes that the scope of piping systems for RI-ISI should be plant-specific, and the staff is not endorsing WCAP-14572 pilot list of systems for generic use. The staff also finds acceptable the discussion of partial scope option which is consistent with guidance provided in RG 1.178 and SRP Chapter 3.9.8 which state that the partial scope option is acceptable as long as it is well defined, and the change in risk due to the implementation of the RI-ISI program meets the guidelines in RG 1.174.

3.2.2 Piping Segments

Section 3.3 of WCAP-14572, Revision 1 provides a definition for piping segments. The approach used to define piping segments was based on the following considerations:

(1) piping failures that lead to the same consequence determined from the plant-specific PRA and other considerations (e.g., loss of a residual heat removal (RHR) train, loss of a refueling water storage tank (RWST), inside or outside containment consequences, etc.)

(2) where flow splits or joins

(3) piping to a point where a pipe break could be isolated (This includes check valves and motor-operated or air-operated valves. No credit is generally given for manual valves however, situations may occur where manual valves can be used to isolate a failure by plant operators and, in these cases, the decision for crediting manual valves is made by the plant expert panel and documented as such.)

(4) Pipe size changes

In defining pipe segments, the possibility of check valves and other isolation valves failing to close is not considered; that is, proper operation of the valves is assumed when defining segment boundaries. The staff notes that this assumption will not have a significant impact on the results, since the probability of a valve failing to close is small (ranging from 10^{-2} per demand for motor-operated valves (MOV) to approximately 10^{-4} per demand for check valves) and the consequences from failure will not change in most instances. In addition, when operator action is credited for the isolation of a pipe break, the valve failure probability will be small when compared to the human error probability, and this combined probability will be subject to a sensitivity study as discussed in Section 3.3 of this safety evaluation report (SER). Finally, the treatment of automatic isolation valves will be clarified as follows (item 9 of Ref. 8):

"Automatic isolation valves are assumed to close if the pipe failure in question would create a signal for the valves to close. Containment isolation valves should be carefully considered for segments which contain the containment penetrations. If the segment consequences are significantly different assuming an automatic and/or containment isolation valve failure, then the piping segment definition should be reviewed and if

necessary, the piping segment should be further combined or subdivided such that the failure of the valve, under pipe failure conditions, would be considered in conjunction with the change in consequences."

The staff finds that the definition of a piping segment, as addressed in Section 3.3 of WCAP-14572, Revision 1 (and subject to the revision noted above) is acceptable since this definition is consistent with the expectations expressed in Section 4.1.4 of RG 1.178 which states that one acceptable approach to divide piping systems into segments is to identify segments as portions of piping having the same consequences of failure in terms of an initiating event, loss of a particular train, loss of a system, or combination thereof. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.2.3 Piping Failure Potential

WCAP-14572 methodology is based on industry experience and the Structural Reliability and Risk Assessment (SRRA) computer code to determine the failure probabilities of piping segments. The staff believes that the purpose of the piping failure probability estimation is to provide a relative estimate of the piping failure potential in order to differentiate the piping segments based on potential failure mechanism and postulated consequences. The relative failure probabilities of piping segments provide insights for use by the expert panel in defining the scope of inspection for the RI-ISI program. Section 3.4 of this SER provides a detailed discussion of the qualification and role of the expert panel.

At its briefing in July 1997, the NRC's Committee to Review Generic Requirements (CRGR) requested that the staff should have a peer review performed with regard to using structural reliability and risk assessment computer codes to estimate the probability of a piping failure. The peer review, performed by Battelle-Columbus, and documented in a letter report (Ref. 10), concluded that the SRRA computer code is technically sound and within the state-of-the-art, and that its application can facilitate risk-informed regulatory decision-making in the area of ISI.

Over the past 3 years, as ASME-Research and the WOG developed methods to perform RI-ISI programs for piping, the staff held public meetings with both groups to develop guidelines for acceptable uses of probabilistic fracture mechanics computer codes. In addition, with the assistance of Pacific Northwest National Laboratory (PNNL), the staff performed independent audit calculations to validate the results of the SRRA computer code.

Computer programs CLVSQ and other SRRA computer codes for RI-ISI, such as LEAKMENU and LEAKPROF, were developed, verified and controlled in accordance with the Westinghouse Quality Management System.

Section 3.5 of WCAP-14572, Revision 1 presents general discussion of failure probability determinations; the details of the methodology, process, and rationale are contained in Supplement 1 to the WCAP-14572. This includes piping failure modes, degradation mechanisms, SRRA models, program input, uncertainties, and calculation of failure probability over time. Piping failure potential was determined based on failure probability estimates from the SRRA software program. This software uses Monte-Carlo simulation to calculate the probability of a leak or break for Type 304 or 316 stainless steel piping or for carbon steel piping.

It is recommended in Section 3.5.2, that known failures at other plants be considered and evaluated for applicability.

Section 3.4 of WCAP-14572, Supplement 1, addresses the treatment of uncertainties in the failure probability assessments. The statistical variations for a number of input parameters are discussed therein. Material properties such as yield strength, ultimate strength, fracture toughness, and tearing modulus are not mentioned, but inputs for these properties are more appropriately addressed in plant-specific applications of the program.

WCAP-14572 methodology involves assigning all significant degradation mechanisms present in the segment to a single weld, and imposing the operating characteristics and environment to that weld. The failure probability developed from the Monte-Carlo simulation of this weld is subsequently used to represent the failure probability of the segment, regardless of the number of welds in the segment, or the length of the segment. WCAP-14572 states that this approximation is appropriate since the same loadings occur across the segment and a single weld failure will fail the segment. WCAP-14572 also states that failures in a piping segment due to the dominating failure mechanisms are correlated, and that the failure probability of the weld subject to the dominating mechanisms is typically several orders of magnitude higher than those without the dominating mechanisms. When more than one degradation mechanism is present, the combination of all significant degradation mechanisms for the segment failure probability should produce a limiting failure probability. The output of the SRRA code is thus best described as a relative estimate of the susceptibility of a pipe segment to failure as determined by the weld material and environmental conditions within the segment. The WOG methodology primarily uses these estimates in the following ways:

- Combine with quantitative risk estimates from the PRA to support the expert panel's classification of segments into LSS or HSS.

- Provide guidance regarding the susceptibility of each segment to failure during the sub-panel's selection of welds to be inspected under the RI-ISI program.

Since the WCAP-14572 methodology involves assigning all significant degradation mechanisms present in the segment to a single weld, and imposing the operating characteristics and environment to that weld, the staff finds the methodology acceptable to estimate pipe segment failure probabilities, i.e., the estimation of relative failure probabilities is sufficiently robust to support categorization of pipe segments by the expert panel when this information is used in conjunction with considerations of defense-in-depth and safety margins to support the RI-ISI change request.

The staff also finds it acceptable that the SRRA code assumes that unstable fractures (ruptures) of piping are governed by the limit load criterion because it meets the limit load criterion used in the ASME Code, Section XI, Appendix H, for unstable fractures. The Log-Normal distributions of flaw aspect ratios are based on the same assumptions used in the pc-PRAISE code, an NRC sponsored code.

The Monte-Carlo method as implemented into the SRRA code is a standard approach which is commonly used in probabilistic structural mechanics codes including the pc-PRAISE code. Importance sampling, again a common and well-accepted approach, increases the

computational efficiency of the Monte-Carlo procedure by shifting the distributions for random variables to increase the number of simulated failures. The magnitude of shift applied to the variables by the SRRA code is relatively modest and is not believed to be sufficient to cause incorrect estimates of failure probabilities. The staff finds the numerical method acceptable because it represents standard probabilistic fracture mechanics techniques, is based on sound, generally accepted principles of solid mechanics, and is consistent with guidance provided in RG 1.178 and SRP Chapter 3.9.8.

WCAP-14572 states that the median values for stresses were set equal to one-half the stress values calculated by ASME Code stress analysis. In the public meeting on September 22, 1998 [item 2, Ref. 8], Westinghouse stated that in most piping stress analyses, dead weight, thermal, and pressure stresses are calculated on the basis of conservative assumptions such as concentrated dead loads, rigid support stiffnesses, conservative design conditions and stress concentration factors. Westinghouse also stated that the next revision of WCAP-14572 will clarify that if piping stress analysis is performed on the basis of realistic rather than conservative assumptions, higher median values and lower uncertainty can be justified and used in the detailed input options. Conditioned upon this change being incorporated into the next revision of WCAP-14572, the staff concludes that the approach for estimating the median values for stresses is acceptable because it is based on assumptions of conservative stresses in common pipe stress analyses and also accounts for situations when realistic, rather than conservative, values of dead load and thermal stresses are used.

In the public meeting on September 22, 1998 [item 3, Ref. 8], Westinghouse stated that the welding residual stresses used in the SRRA code are consistent with the pc-PRAISE code. Because of conservatism in applying these stresses in the SRRA code, the residual stresses are truncated at a maximum value of 90% of the material flow stress. Westinghouse also stated that the next revision of WCAP-14572 will provide basis for estimating the residual stresses to be used in the SRRA code. The staff finds the estimation of residual stresses to be acceptable because the conservatism that the residual stress is assumed to be constant through the weld wall and around the circumference, and no relaxation of residual stress is assumed for an initial fabrication flaw justifies the assumption that the yield strength of the weld is assumed to be 90% of the flow stress in the SRRA code for RI-ISI. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 as described above.

In the public meeting on September 22, 1998 [item 4, Ref. 8], Westinghouse stated that industry experience has shown that axial cracks which could initiate from longitudinal welds are not a serious concern and have a low probability of occurrence because of the normal pressure and temperature ranges associated with nuclear operating plants. ASME Code Case N-524 was written to eliminate the requirement to examine longitudinal welds beyond the region of intersection with circumferential welds. The staff concludes that this approach is acceptable to address the axial cracks that could initiate from longitudinal welds, conditioned on Westinghouse revising WCAP-14572 [item 4, Ref. 8] to state that in the rare situation that a longitudinal weld or nonstandard geometry would need to be evaluated, the failure probability should be estimated by other means, such as expert opinion or advanced modeling.

The PRODIGAL program is used to calculate the number of flaws per weld length near the inner surface of the pipe. The staff concludes that this treatment of near-surface flaws is adequate and acceptable because all near-surface flaws are assumed to be inner surface breaking flaws,

the stress intensity factor for the near-surface flaws are conservatively calculated in the SRRA fracture mechanics models, and the flaw density used for the failure probability calculation is not reduced to eliminate the effect of flaws that are not actually surface flaws. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above as stated by Westinghouse in the public meeting on September 22, 1998 [item 4, Ref. 8].

The CLVSQ program uses a simplified correlation to calculate leak rates. The staff finds the leak rate model to be acceptable since the accuracy of the correlation for fatigue type cracks is estimated to be within 25% and was judged to be acceptable by the ASME Research Task Force. PNNL's studies with pc-PRAISE also showed that the large leak and break probabilities were relatively insensitive to the actual value of the detectable leak rate in the range of 0.3 to 300 gpm [item 5 (c), Ref. 8].

The staff had identified an open item that WCAP-14572, Revision 1, does not identify the value that is used for the high-cycle fatigue stress for the 1-inch pipe size. Westinghouse clarified in the public meeting on September 22, 1998 [item 6, Ref. 8], that the vibration input for 1-inch pipe size is an input parameter determined by the SRRA user and an insert will be added in WCAP-14572 to provide guidelines for the SRRA user. A correction factor is applied to this stress to obtain the fatigue stress for other pipe sizes. The staff finds this approach to be acceptable since it specifies that the simplified input parameter is the peak-to-peak vibratory stress range in ksi corresponding to a one-inch pipe size. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

Figure 4-2 of WCAP-14572, Revision 1, Supplement 1, graphically compares SRRA model predictions with industry plant data relative to the probability of violating minimum wall thickness criteria because of flow-accelerated corrosion wastage. The staff had expressed a concern (Ref. 9) that the graph indicates that the SRRA model tends to over-predict the failure probability early in plant life and to under predict later in life. In the public meeting on September 22, 1998 [item 7 (a), Ref. 8], Westinghouse explained that the minor over-prediction early in life is attributable to lower plant startup capacity factors (fraction of time at full power and flow), while the minor under-prediction later in life is attributable to higher capacity factors during this more mature period of plant operation. The staff finds this response acceptable since the industry observed failure rates due to wastage are within a factor of 2 to 3 of the SRRA calculated values even though the calculation was based upon data averaged values of pipe size and wall thickness.

Supplement 1 to WCAP-14572 provides information on assumptions made in the SRRA wall thinning model. In the public meeting on September 22, 1998 [item 7 (b), Ref. 8], Westinghouse stated that the next revision of WCAP-14572 will provide guidance for material wastage potential consistent with Ref. 11. The staff concludes that the guidance for estimating the material wastage potential is acceptable since, if material wastage rates are high enough to proceed through the pipe wall, the probabilities of small leak, large leak and break are all calculated to be the same. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above. In addition, the acceptance is limited to this application, i.e., development of a risk-informed ISI program. As noted elsewhere, the licensees' augmented programs for erosion-corrosion will not be changed as a result of this alternative, and the staff is not endorsing the SRRA code for application in such augmented programs.

The staff had identified an open item that WCAP should provide guidance for the analyst on the SRRA code limitations for complex geometries and guidance for effective use of the code in such applications. In the public meeting on September 22, 1998 [item 12, Ref. 8], Westinghouse stated that the SRRA piping models only apply to standard piping geometry (circular cylinders with uniform wall thickness). Westinghouse further stated that a limitation on the use of nonstandard geometry will be added in the next revision of WCAP-14572. The staff finds this clarification of the code limitation to be acceptable. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

The staff had also indicated that WCAP should specify the level of training and qualification that the code user needs to properly execute the SRRA code. In the public meeting on September 22, 1998 [item 13, Ref. 8], Westinghouse indicated that the next revision of WCAP-14572 will state that to ensure that the simplified SRRA input parameters are consistently assigned and the SRRA computer code is properly executed, the engineering team for SRRA input should be trained and qualified. The revised WCAP will also list the topics covered in this training as described in the September 22, 1998, public meeting [item 13, Ref. 8]. The staff finds the level of training and qualification that the code user needs to properly execute the SRRA code to be acceptable since it includes training on overall risk-informed ISI process , and how SRRA calculated probabilities are used in the piping segment risk calculation. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

It was the staff's understanding that the existing correlation for leak rates are limited to pressurized-water reactors (PWR) reactor coolant system (RCS) conditions. The staff had indicated (Ref. 9) that Westinghouse should clarify whether the SRRA code can be applied to boiling-water reactors (BWR) and justify the applicability of the correlations used to calculate leak rates under BWR operating conditions. In the public meeting on September 22, 1998, Westinghouse stated that the existing correlations for leak rates can be used for other plant conditions beyond the RCS and that the SRRA code can be applied to BWRs; however, care must be exercised in applying this approach to BWR piping systems, particularly those subjected to intergrannular stress corrosion cracking (IGSCC). In addition, Westinghouse indicated that WCAP-14572 will be revised [item 5(d), Ref. 8] to provide guidance on addressing stress corrosion cracking. The staff finds the response acceptable since most piping susceptible to stress corrosion cracking (SCC) is also subject to fatigue loading, such as normal heat up and cool down, and the leak rate correlation for fatigue type cracks was conservatively assumed for the CLVSQ Program. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

The staff had identified an open item that WCAP should describe how proof testing is addressed in the SRRA calculations. In the public meeting on September 22, 1998 [item 14, Ref. 8], Westinghouse stated that the effect of proof testing on the segment risk ranking and categorization would be very small and slightly conservative. Westinghouse also indicated that the next revision of WCAP-14572 will clarify that SRRA models in LEAKPROF do not take credit for eliminating large flaws, which would fail during the pre-service hydrostatic proof test, even though this is allowed as an input option in pc-PRAISE. The staff concludes that the approach for addressing proof testing is acceptable because Westinghouse has demonstrated that the effect of proof testing on the segment risk ranking and categorization would be very small and slightly conservative. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

Before issuing this SER, the staff had identified an open item that the probability of detection curves used in calculations need to be justified for the material type, inspection method, component geometry, and degradation mechanism that apply to the structural location being addressed. In the public meeting on September 22, 1998 [item 15 (a), Ref. 8], Westinghouse stated that the default input values for the probability of detection (POD) curves are consistent with the default input values for pc-PRAISE. The revised WCAP will emphasize that the SRRA code user must ensure that the specified input values for POD are appropriate for the type of material, inspection method, component geometry, and degradation mechanism being evaluated. The staff finds this response acceptable since POD curves are consistent with the default input values for pc-PRAISE code which has been validated and accepted by the staff for various applications. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

Before issuing this SER, the staff had identified an open item that Westinghouse should expand the code documentation to provide additional guidance for selecting the input for the calculation. In the public meeting on September 22, 1998 [item 15 (b), Ref. 8], Westinghouse stated that the next Revision of WCAP-14572, Supplement 1, will provide detailed guidelines for simplified input variables and any associated assumptions that could be important in assigning the input values for the SRRA code. WCAP-14572 will also state that if more than one degradation mechanism is present in a given segment, the limiting input values for each mechanism should be combined so that a limiting failure probability is calculated for risk ranking. The staff finds the guidance in item 15 (b), Ref. 8 to be acceptable because it provides sufficient guidance for the code user for selecting input parameters. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.2.4 Consequence of Failure

The consequences of the postulated pipe segment failures include both direct and indirect effects of each segment failure. The direct effects include failures that cause initiating events or disable system trains or entire systems as a result of the loss of flow paths or loss of inventory, and the possible creation of diversion flow paths. Indirect effects include spatial effects, such as flooding, water spray, pipe whip, and jet impingement. WCAP-14572 methodology relies on the use of PRA models and results to gain insights into the potential direct and indirect consequences of pipe failures. Plant walkdowns are also an integral part of the methodology. The staff finds the general guidance provided in WCAP-14572 to determine the direct and indirect consequence of segment failure to be acceptable because it is comprehensive and systematic, and should produce a traceable analysis. WCAP-14572 does not include a detailed discussion of the specific assumptions to be used to guide the assessment of the direct and indirect effects of segment failures. For example, although diversion of flow is included as a direct effect, there is no guidance for determining whether a flow would be sufficiently large to fail a system function. Similarly, WCAP-14572 does not provide clear guidance for calculating flooding effects with regard to the required modeling of flood propagation pathways, modeling of flood growth and mitigation, and assumptions for the failure of critical equipment within a flood zone (e.g., if electro-mechanical components must be submerged before failure, etc.). The staff finds that specific assumptions regarding the direct and indirect effects of pipe segment failure should be developed by the individual licensees and should form part of the onsite documentation. A revision to WCAP-14572 (see item 8 (e) in Ref. 8) will require that details from

the consequence evaluation be maintained onsite for potential NRC audit.

WCAP-14572 methodology recommends considering a spectrum of different size breaks (i.e., failure modes) in every segment. The failure modes considered are the small leak, the disabling leak, and a full break, as discussed in Section 3 of Supplement 1. Failure probability for each of these modes typically decreases as the size of the break increases. WCAP-14572 also defines the direct and indirect effects to be evaluated for each postulated failure mode. The staff finds that the association between failure mode and effects is reasonable when compared to previous results and findings from PRAs of internal flooding events.

In section 3.4.2 of WCAP-14572 it is stated that the indirect effects of a pipe whip need not include the rupture of other piping of equal or greater size, but it should be assumed that a through-wall crack will develop in a line that is impacted by a whipping pipe of the same size. In Ref. 8, Westinghouse stated that the bases for these assumptions are found in Ref. 13 and Ref. 14. These references also provide justification for WCAP-14572 guidance on the location of circumferential and longitudinal breaks in high energy piping runs. In accordance with item 10 of Ref. 8, Ref. 13 and Ref. 14 will be added to the WCAP-14572, and cited appropriately in the text. The staff finds that the bases found in Ref. 13 and Ref. 14 to be acceptable because they represent established and commonly accepted industry practices. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.3 Probabilistic Risk Assessment

The requirements of a PRA and the general methodology for using PRA in regulatory applications is discussed in the guidelines in RG 1.174. RG 1.178 provides guidance that is more specific to ISI. It is expected that licensees who wish to apply the WCAP-14572 methodology to an RI-ISI program will also conform to the RGs 1.174 and 1.178 guidelines for PRA quality, scope, and level of detail.

In July 1997, at staff briefing of the CRGR on draft RG 1.178, CRGR suggested that a peer review be performed of the use of PRA methods to support RI-ISI. The methodology proposed in RG 1.178 is similar to that found in WCAP-14572. The peer review, performed by Brookhaven National Laboratory (BNL), and documented in a letter report (Ref. 12), concluded that the PRA approach is technically sound and within the state-of-the-art, and that the approach can facilitate risk-informed regulatory decisionmaking in the area of ISI.

WCAP-14572 does not prescribe the incorporation of pipe segment failure events into the PRA model. Instead, the core damage frequency (CDF)/large early relief frequency (LERF) for each segment is determined by the use of surrogate events (i.e., initiating events, basic events, or groups of events) already modeled in the PRA with failures that are representative of the effects of the piping segment failure. By setting the appropriate surrogate events to a failed state in the PRA and by re-quantifying the PRA, the impact of the pipe segment failure can be estimated. The staff finds this process acceptable as long as the truncation limits used in the baseline calculations are maintained and the model is re-quantified. If a pre-solved cutset/scenario model is used instead of re-quantifying the baseline model, the application should include justification as to why the truncated model still produces reasonable results given that the equipment is assumed to be failed.

The segment failure probability/rate is combined with the results of the risk calculation as described in Equations 3-1 to 3-10 of WCAP-14572, Revision 1. The results are subsequently combined into a total piping segment CDF (or LERF). The staff recognizes that the WCAP equations are approximations for segment failures which do not trip the plant and that are discovered before an unrelated plant trip. Following the discovery of such a rupture, the likely operator action would be to isolate the break and to decide whether to shutdown or to continue plant operation. In some cases, the break may disable equipment required by the technical specifications and plant operation will be governed by allowed outage time (AOT). If the decision is made not to shut down the plant, the licensee would presumably realign the affected systems to facilitate repairs. If the decision is made to shut down the plant, the licensees may realign the systems to provide more robust mitigating function capabilities during the shutdown process, or may simply begin a controlled plant shutdown. In all cases except the long AOT scenario, the degraded condition would only be present during a relatively short time span. Furthermore, a pipe segment rupture is an unusual event and the operations staff would be very aware of the degraded functions and would be prepared to actively intervene if necessary. The staff finds the assumption that short AOT and controlled shutdown risk are minor contributors compared to risks associated with segment failure following an unrelated transients acceptable because of the short exposure time and the heightened awareness by the plant staff.

Short exposure time and heightened plant staff awareness may not, however, be a reasonable assumption if there is a long AOT. In response to staff comments, Westinghouse indicated that in a future revision to WCAP-14572 [item 18, Ref. 8], Equation 3-8 will be modified such that, for systems in which outage times are approximately the same order of magnitude as the test interval (T_t), e.g., approximately $\frac{1}{4}T_t$, the contribution attributed to maintenance unavailability (expressed as FR_{PB} * AOT) will be added to the total component unavailability.

The staff notes that the description associated with equation 3-5 on page 97 of the WCAP is not an appropriate characterization of the "CCDF" variable in the equation. The equation estimates what the WCAP refers to as a "Conditional Core Damage Frequency" (CCDF) to characterize the risk due to pipe failures that do not cause an initiating event but only fail mitigating systems. The staff believes that the desired quantity is not the conditional core damage frequency given a pipe break as stated, but rather the increase in the core damage frequency when the pipe break probability is changed from zero to unity. This change is multiplied by the pipe break failure probability to obtain the core damage frequency due to the pipe break. With this change in definition (e.g., CCDF as Change in Core Damage Frequency) of the result being calculated by the equation, the equation is correct and acceptable.

The staff notes that Equation 3-8 on page 99 is used to characterize several slightly different failure modes of piping segments. For failure modes where the pipe is continuously degrading and eventually reaches the point that transient or additional stresses associated with a demand following an initiating event would cause the pipe to fail, the equation corresponds to the normal standby failure estimate (e.g., the pipe integrity has failed but the failure only becomes apparent on demand). If the segment does not continuously degrade, but the strength is degraded slightly on each test demand, the equation is also a valid approximation. If the pipe does not degrade, but there are variations in the demand stress, the equation underestimates the failure probability by a factor of two. The staff finds the approximation acceptable since it is valid for the most likely failure modes, and produces a reasonable approximation for the other failure mode.

The staff finds that the methodology will yield results of commensurate precision with the segment failure probabilities and which, after review by the expert panel, can be used to support safety significance determination.

3.3.1 Evaluating Failures with PRA

The staff finds that the discussion in Section 3.6.1 of WCAP-14572, Revision 1, concerning the evaluation of CDF/LERF using surrogate components needs clarification with regard to the incorporation of indirect consequences associated with pipe segment failures. Since WCAP-14572, Revision 1, does not explicitly state that all components subject to a harsh environment, jet impingement, pipe whip, etc., initiated by a pipe segment failure should be failed in the PRA model evaluation, individual applications utilizing WCAP-14572 methodology must assume failure of this equipment in the risk evaluation, or provide justification as to why failure is not assumed in order to be considered an acceptable implementation of WCAP-14572 (e.g., the component is environmentally qualified to the conditions expected from the pipe failure event).

For some initiating events and plant operating modes, the scope of the available plant-specific PRA models may not be sufficient to estimate the impact of a pipe segment failure. For example, some PRAs may not model fires, seismic or other external events, and the shutdown mode of operation to the level of detail required to estimate relative risk importance or risk impact. For these cases, the impact of failure of each pipe segment on risk must then be developed and incorporated in the decision-making process by an expert panel. WCAP-14572 provides sample expert panel worksheets that include a listing and discussion of the safety-significant functions a system must perform. The expert panel is expected to consider the importance of these functions for scenarios not modeled in the PRA so that the categorization of safety significance of the pipe segments reflects all plausible accident scenarios. Since the text in WCAP-14572 does not discuss system functions and their use by the expert panel, individual RI-ISI applications must address this issue in order to be considered an acceptable implementation of WCAP-14572.

3.3.2 Use of PRA for Categorizing Piping Segments

Based on quantitative PRA results which assume no credit for ISI, risk reduction worth (RRW) and risk achievement worth (RAW) measures are developed for each pipe segment as described in Equations 3-11 and 3-12 of WCAP-14572. The RRW calculates the current contribution of the segment failure to risk and the RAW calculates the potential change in risk associated with the failure of the pipe segment. Use of these measures provides useful insights to the integrated decision-making process. The staff finds that the use of quantitative models which assume no credit for ISI is appropriate for the determination of the safety significance of pipe segments because one of the goals of the RI-ISI program is to target the inspection of those elements where inspection will be most efficient. If a pipe segment has one or more welds inspected under an augmented inspection program, WCAP-14572 methodology specifies that the representative weld failure probability is calculated assuming credit for ISI. The use of quantitative models which credit ISI for segments inspected under the augmented program is

appropriate since the augmented program inspection is maintained in the RI-ISI process.

WCAP-14572 recommends that pipe segments with RRW greater than 1.005 should be categorized as HSS while the segments with RRW values between 1.001 and 1.004 should be identified for additional consideration by the expert panel. The staff recognizes the utility of the suggested RRW guidelines and finds that these suggested values may be used for initial screening. WCAP-14572 does not provide guidelines for the RAW values for classification of safety significance. Instead, WCAP-14572 suggests that these values should be generated and supplied to the expert panel for consideration. The staff finds that the RAW values, or some other measure of the consequence of segment failure, provides a valuable input to the decision making process. The expert panel should be aware of the implications of high RAW values (or other consequence measure) so that their decisions are made with a full understanding of the severity of the consequences of each segment's rupture. The appropriateness of the RRW guidelines and use of the RAW values should be documented as part of the licensee's categorization process and should be assessed on a plant-specific basis within the framework of the proposed ISI program and based, in part, on the risk impact from the application.

An integral part of the categorization process is the expert panel which makes a final determination of the safety significance of each pipe segment. The expert panel considers pipe segment characteristics (e.g., Table 3.6-9 of WCAP-14572, Revision 1), the system characteristics (e.g., Table 3.6-12 of WCAP-14572, Revision 1), the risk-related information in the form of relative pipe segment importances and consequences of pipe failure, and information not available from the risk analyses such as the importance of the pipe for mitigating unquantified events (shutdown, external events, etc.). In addition, guidance to be added to Section 3.6.3 of WCAP-14572 [item 8(c), Ref. 8] will ensure consistent application of the expert panel process. Section 3.4 of this SER provides a detailed discussion of the qualification and role of the expert panel. The staff finds that in the categorization of pipe segments, the use of an expert panel (as documented in Section 3.6.3 of WCAP-14572) to combine PRA and engineering information (as described in example Tables 3.6-9 and 3.6-12) is acceptable and necessary. The staff finds the process acceptable since it meets the intent of the integrated decision-making process guidelines discussed in RGs 1.174 and 1.178, in that engineering and risk insights (both qualitative and quantitative) are taken into consideration in identifying safety significant piping segments. The staff notes that the expert panel's records must be retained on site and available for NRC staff audits. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.3.2.1 Sensitivity to Modeled Human Actions

Operator actions to isolate a break and mitigate its immediate consequences are credited in the RI-ISI analysis. For example, operator action to close an MOV to stop the loss of water from a break can be credited, if this action is shown to be feasible. WCAP-14572 methodology recommends that two sets of calculations be performed, one assuming all such actions are successful and another assuming that all such actions fail. The RRW and RAW measures are calculated for these different assumptions and if the RRW is greater than 1.005 for the CDF or LERF calculations with or without operator action the segment is classified HSS. If any RRW is between 1.005 and 1.001, safety significance considerations are reviewed and the safety significance determined during the expert panel deliberations. The staff finds it acceptable to

use sensitivity studies to bound the possible impact of operator actions since these sensitivity calculations may point to areas where credit for recovery actions plays a major role in the classification of pipe segments (and where licensee commitment to these actions is important, or dependence on these recovery actions can be lessened).

In addition to operator recovery actions, the modeling of human actions can affect the RI-ISI process in another way. Specifically, choosing a surrogate PRA component to represent the system effects of a pipe failure in a segment must include consideration of how the surrogate component is modeled in the PRA, including the modeling of recovery actions for the component. To emphasize this consideration when choosing surrogate components, the following will be added to a future revision of WCAP-14572 [item 8 (d) of Ref. 8]:

> "When choosing a surrogate component, care must be taken to account for the ways in which the component has been modeled in the PRA, including recovery actions which may have been modeled to restore the operability of the component. If the recovery action was determined to be inappropriate for the postulated consequence given a piping failure, the recovery action basic event should also be failed with a probability of 1.0."

The staff finds the above addition to be acceptable since operator recovery actions that are no longer feasible as a result of a flood, will no longer be credited. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.3.2.2 Sensitivity to Segment Failure Probability

WCAP-14572 includes an evaluation in which the impact of the variation in the segment failure probabilities on the safety significance determination is investigated. The analysis was based on assigning a range factor to the pipe failure probabilities. The staff finds that this study is useful and should be performed on a plant-specific basis for RI-ISI applications so that the impact of the variation of the pipe failure probabilities on the safety significance classification process can be evaluated.

As part of the staff's review of the WCAP methodology, independent audit analyses were performed by PNNL to estimate the uncertainties in the calculated failure probability for a piping segment. Highlights of the uncertainty studies are documented in NUREG-1661 (Ref. 15). The results from the uncertainty studies are illustrated in Figure-1 and summarized below:

1. The upper bound curve was based on the largest of the 100 failure probabilities calculated from the 100 pc-PRAISE runs for each given cyclic stress level.

2. The largest uncertainties are for those cases that have very low values of calculated failure probabilities. The uncertainties decrease with increasing failure probabilities.

3. The categorization of piping segments as high- and low-safety-significant is a function of the degradation mechanism and consequences. "Inactive" versus "active" degradation mechanisms result in significant variation in failure probabilities. This variation renders the impact of the large uncertainties for components with low failure probabilities as having a relatively small impact on the categorization. The effects of uncertainties on component categorization can be accounted for through numerical evaluations, such as Monte Carlo

analyses.

4. The calculations for components with very low failure probabilities are particularly sensitive to the tails of the distributions assumed for input parameters such as flaw depths and crack growth rates. The large uncertainties in the calculated failure probabilities are a direct results of the fact that the tails of these input distributions are based on extrapolations from actual data.

5. Failure rates for components with high calculated failure probabilities can be assessed for consistency with plant operating experience and with industry data bases on reported field failures. The ability to make such comparisons helps to minimize the uncertainties in the calculated probabilities.

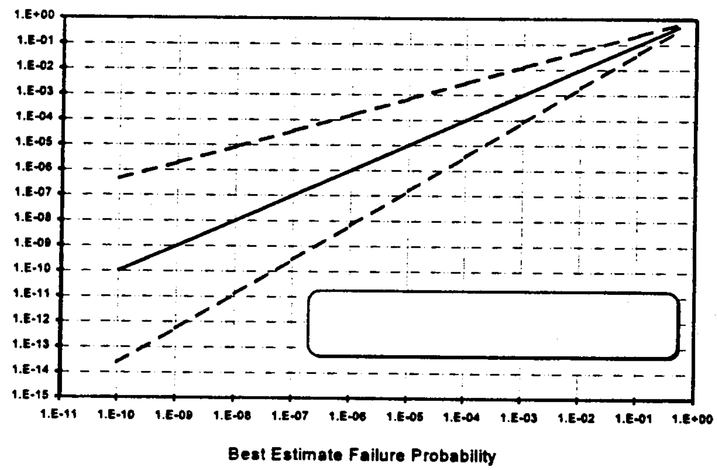

Figure - 1 Uncertainty Bounds Related to Values of Calculated Failure Probabilities

To ensure that the potential impact of uncertainties is adequately addressed in the categorization of piping segments, Westinghouse committed to add the following as part of a future revision to WCAP-14572 [item 19, Ref. 8]:

> "In addition to the sensitivity studies described above, a simplified uncertainty analysis is performed to ensure that no low safety significant segments could move into the high safety significance category when reasonable variations in the pipe failure and conditional CDF/LERF probabilities are considered. The results of the evaluation along with other insights are provided to the plant expert panel."

The staff finds that the sensitivity studies as proposed by WCAP-14572 (and as amended by the above addition) would address model uncertainty in terms of pipe failure probabilities, and would ensure that pipe segment categorization is robust. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.3.3 Change in Risk Resulting from Change in ISI Programs

To estimate the change in risk from the implementation of the RI-IST program, WCAP-14572 methodology utilizes the SRRA code to provide a quantitative estimate of the relative susceptibility of pipe segments to failure as determined by the weld material and environmental conditions within the segment. Different weld failure probabilities are calculated depending on whether the weld is inspected or not. The methodology credits the reduction in weld failure probability attributable to ISI at the segment level. If one or more welds within a segment are inspected under the current Section XI program or the RI-ISI program, the selected weld failure probability including credit for ISI is assigned to the segment. That is, the segment failure probability will not change as a result of any changes in the inspection strategy applied to the welds within a segment. If one or more welds were inspected under the Section XI program, but no welds will be inspected under an RI-ISI program, the segment failure probability will increase. If no welds were inspected under the Section XI program, but one or more welds will be inspected under the RI-ISI program, the segment failure probability will decrease. If one or more welds within a segment are inspected in the augmented program, the selected weld failure probability including credit for the augmented program is assigned to the segment. For a selected pipe segment where at least two separate inspections are being performed (one for the primary failure mechanism which is addressed by an augmented program, and other inspection(s) performed under the Section XI program or the RI-ISI program, so that the secondary mechanism is addressed), a factor of three improvement in the failure probability is credited.

The staff finds the above process acceptable, but recognizes that this process underestimates risk reductions arising from changing inspection locations from a weld subject to no degradation mechanism to another with an identified degradation mechanism. It also underestimates risk increases arising from the reduction in the number of welds inspected within each segment. The staff expects that the targeting of inspections to degradation mechanisms should yield relatively large risk reductions, while the reduction in the number of inspections within a segment will yield a larger number of smaller risk increases. However, as discussed in Section 3.2.3 of this SER, the increase in risk resulting from a reduction in the number of inspections should be minimal since WCAP-14572 methodology will characterize the failure probability of a segment by combining the failure probabilities of the dominant degradation mechanisms in that section.

In determining whether the change in CDF and LERF associated with WCAP-14572 methodology is acceptable, the following factors were also considered; the statistical evaluation used to develop an initial estimate of the number of welds to inspect, and the four criteria for evaluation of results found in Section 4.4.2 of WCAP-14572. These are further discussed below.

To ensure that a target leak rate is met with a stated level of confidence, the statistical evaluation methodology proposed in WCAP-14572 uses the probability of a flaw, the conditional

probability of a leak, and a target leak rate to determine the minimum number of welds to inspect. In discussions with the staff, Westinghouse stated that, in controlling the frequency of pipe leaks, the pipe break frequency (which drives the safety significance classification) is also controlled. This is supported by the pilot WCAP RI-ISI application, which reported that the conditional probability of a pipe break is sufficiently small when compared to the conditional leak probability, and that the level of confidence that the target leak frequency is not exceeded is also the confidence that the pipe break frequency is not exceeded. WCAP-14572 methodology thus provides a systematic evaluation of the required number of inspections that is acceptable for the RI-ISI program, and confidence that the failure likelihood of high safety significant piping segments will not increase above those values used to support the finding.

WCAP-14572 provides guidelines for evaluating the change in plant and system-level risk resulting from changes to the ISI program. The first guideline suggests the addition of examinations until at least a risk neutral change is estimated. The second guideline suggests that the risk-dominant pipe segments within systems which dominate the estimated risk (e.g., greater than 10% of the total) should be reevaluated to identify where additional examinations may be needed so that the overall risk for these systems could be reduced. The third guideline suggests that, for systems where risk increases are identified, additional examinations may be necessary to minimize the risk increase (to less than two orders of magnitude below the RI-ISI CDF/LERF for that system and less than a 10^{-8} CDF increase or a 10^{-9} LERF increase). The staff finds that these WCAP guideline are consistent with the guidance in RGs 1.174 and 1.178 which state that risk increases (if any) resulting from a proposed change should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

In summary, the staff finds that, although the calculation of the change in risk (CDF/LERF) will not precisely estimate the magnitude of the change, the calculation can illustrate whether the resulting change will be a risk increase or a risk decrease. Using sensitivity studies, the quantitative results can be shown to be robust in terms of credit for operator actions and pipe segment failure probability. By utilizing plant and system-level criteria as discussed above, the risk from individual system failures will be kept small and dominant risk contributors will not be created. When applied as part of an integrated decision-making process, the staff finds that the analyses, results, and decision criteria associated with the determination of segment safety significance and subsequent change in risk estimates provide reasonable assurance that the change in the ISI program would result in a total plant risk neutrality, risk decrease, or a small risk increase that will be consistent with staff guidelines found in RG 1.174. For full scope RI-ISI programs, such as the one performed for Surry Unit 1, the staff anticipates the program to be risk neutral or result in a risk reduction.

3.4 Integrated Decisionmaking

RG 1.178 and SRP Chapter 3.9.8 guidelines describe an integrated approach that should be utilized to determine the acceptability of the proposed RI-ISI program by considering in concert the traditional engineering analysis, risk evaluation, and the implementation and performance monitoring of piping under the program.

In the WCAP-14572 approach to integrated decisionmaking, conventional fracture mechanics analysis methods are combined with Monte-Carlo probabilistic simulations to determine failure

probabilities for the pipe segments, as discussed in Supplement 1 to WCAP-14572, Revision 1. These failure probabilities are used together with the results of consequence evaluations to characterize the conditional risk associated with the failure of each segment, as discussed in Section 3.6 of WCAP-14572. Specifically, section 3.6 explains how this information is integrated with deterministic considerations and an expert panel evaluation to categorize pipe segments as either LSS or HSS. Section 3.7 of WCAP-14572, Revision 1, explains how the results of this risk-ranking process are used in selecting structural elements for examination.

An integral part of the RI-ISI process is the expert panel which makes a final determination of the safety significance of each pipe segment. The expert panel is responsible for the review and approval of all risk-informed selection results by utilizing their expertise and past experience in inspection results, industry piping failure data, relevant stress analysis results, PRA insights, and knowledge of ISI and nondestructive examination techniques. The RI-ISI expert panel should include expertise in the following areas:

- PRA
- Plant Operations
- Plant Maintenance
- Plant Engineering
- ISI
- Nondestructive Examination
- Stress and Materials Engineering

Section 3.6.3 of WCAP-14572, Revision 1, provides details of the WOG expert panel process. Item 8(c) of Ref. 8 provides further details on the role of the expert panel to evaluate the risk-informed results and make a final decision by identifying HSS segments for ISI. Item 8(c) of Ref. 8 also states that segments that have been determined to be HSS should not be classified lower by the expert panel without sufficient justification that is documented as part of the program and that the expert panel should be focussed primarily on adding piping segments to the higher classification.

The expert panel evaluations are an established part of the Maintenance Rule implementation and their use in risk-informed applications is well established. The staff finds that in the categorization of pipe segments, the use of an expert panel (as documented in Section 3.6.3 of WCAP-14572) to combine PRA and engineering information (as described in example Tables 3.6-9 and 3.6-12) is acceptable and necessary. In addition, guidance to be added to Section 3.6.3 of WCAP-14572 [item 8(c), Ref. 8] will ensure consistent application of the expert panel process. The staff finds the process acceptable since it meets the integrated decision-making process guidelines discussed in RG 1.174 and SRP Chapter 1.178, in that engineering and risk insights (both qualitative and quantitative) are taken into consideration in identification of safety significant piping segments. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

3.4.1 Selection of Examination Locations

At its July 1997 briefing, CRGR requested that the staff should have a peer review performed to assess the use of Perdue-Abramson statistical model to determine the number of elements to be inspected within a piping segment. The contractor performing the peer review in this area (Los Alamos National Laboratory(LANL)) concluded (Ref. 16) that the Perdue-Abramson method is a statistically sound method for use in determining the number of welds to be inspected in an RI-ISI program in order to ensure that a specified target leak frequency is not exceeded at the pre-specified confidence level of 95%. LANL further stated that although other sampling schemes could be used (such as classical and/or Bayesian double or sequential sampling schemes), the Perdue-Abramson model is capable of providing the desired confidence or assurance.

Section 3.6.1 of WCAP-14572 addresses evaluation of the classification of piping segments, using sensitivity studies to demonstrate whether changes in assumptions or data can affect these classifications. Piping systems at Millstone Unit 3 and Surry Unit 1 were considered in these studies. Operational insights are addressed in Section 3.6.2 of WCAP-14572, which indicates that information obtained from plant operation and maintenance experience is used to identify piping segments having a history of design or operating issues. Section 3.6.3 states that an expert panel reviews and approves the final classification of piping segments on the basis of their expertise and insights as discussed in Section 3.4. A discussion of the risk ranking process is provided in Sections 3.6.4 and 3.6.5 of WCAP-14572.

Sections 3.7.1 and 3.7.2 of WCAP-14572 address the criteria used to determine the number of structural elements selected for examination, consistent with the safety significance and failure potential of the given pipe segment. The RI-ISI program includes examinations of HSS elements contained in Regions 1 and 2 of the element selection matrix (Figure 3.7-1 of WCAP-14572). By the WCAP-14572 selection process, 100% of the susceptible locations (Region 1A) are examined. Elements in Regions 1B and 2 are generally subject to a statistical evaluation process such as the Perdue Model.

The Perdue Model is intended to be used on highly reliable piping to establish a statistically relevant sample size and verify the condition of the piping. In cases where an active degradation mechanism exists, particularly where there is an ongoing augmented program, it is inappropriate to use the Perdue Model for element selection. In these cases, the expert panel must apply other rationales for selecting the number of elements to examine. At Surry, the licensee selected certain elements to address a secondary degradation mechanism and reduce the delta risk compared to current Section XI ISI. In other cases, elements were selected to address defense in depth considerations. As discussed in the public meeting on September 22, 1998 [page 274, Ref. 8], Westinghouse indicated that additional guidance would be added in Section 3.7 of WCAP-14572 to address sample size selection in cases where the Perdue Model could not be applied to state that "additional rationale must be developed when a statistical model cannot be applied to determine the minimum number of examination locations for a given segment."

The staff finds the methodology to determine the number of elements selected for examination to be acceptable since, all HSS segments with known degradation mechanisms will be subject to 100% examination, HSS segments with no known degradation mechanism will be sampled for examination on a sound statistical basis to ensure that a specified target leak frequency is not

exceeded at the pre-specified confidence level of 95%, LSS segments with known degradation mechanisms will be subject to examination in accordance with the licensees defined program, and the final scope of examination will result in a change in risk consistent with RG 1.174 guidelines. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above [page 274, Ref. 8].

3.4.2 Examination Methods

Licensees who wish to apply the WCAP-14572 methodology to an RI-ISI program must conform to the guidelines in RG 1.178 for examination and pressure test requirements. Examination methods and personnel qualification must be in accordance with the ASME Section XI Code Edition and Addenda endorsed by the NRC through 10 CFR 50.55a. For inspections outside the scope of Section XI (e.g., EC, IGSCC) the acceptance criteria should meet existing regulatory guidance applicable to those programs.

The objective of ISI and ASME Section XI are to identify conditions (i.e., flaw indications) that are precursors to leaks and ruptures in the pressure boundary that may impact plant safety. Therefore, the RI-ISI program must meet this objective to be found acceptable for use. Further, since the risk-informed programs is predicated on inspection for cause, element selection should target specific degradation mechanisms.

WCAP-14572, Revision 1, specifies that inservice examinations and system pressure tests are to be performed in accordance with Section 4 of WCAP-14572 which should meet the requirements contained in Section XI of the ASME BPVC Code Edition and Addenda specified in the Owner's current ISI program except where specific references are provided that add supplemental requirements, specify other Code editions and addenda, or recommend/require the use of ASME Code Cases. The examination methods for HSS piping structural elements, specified in Table 4.1-1 of WCAP-14572 are taken directly from Code Case N-577, Table 1. As an alternative to Table 4.1-1, additional guidance for the selection of examination methods is provided in Table 4.1-2 of WCAP-14572, which contains suggested examination or monitoring methods consistent with the configuration of the structural element and the postulated failure mode. This guidance is subject to approval by the Authorized Nuclear Inservice Inspector (ANII) under the requirements of Paragraph IWA-2240 of ASME Section XI. Consistent with RG 1.178 guidelines, all ASME Class 1, 2, and 3 piping systems must continue to receive a visual examination for leakage in accordance with the applicable pressure test requirements of ASME Section XI as endorsed by 10 CFR 50.55a.

3.5 Implementation and Monitoring

The objective of this element of RGs 1.174 and 1.178 is to assess performance of the affected piping systems under the proposed RI-ISI program by implementing monitoring strategies that confirm the assumptions and analysis used in developing the RI-ISI program. To satisfy 10 CFR 50.55a(a)(3)(i), implementation of the RI-ISI program (including inspection scope, examination methods, and methods of evaluation of examination results) must provide an adequate level of quality and safety. The plant-specific application process is covered in Section 5 of WCAP-14572, which provides the framework for applying the risk-informed methods to a

specific plant for the ISI of piping.

Considering that the implementation of the proposed RI-ISI program will greatly reduce the number of examinations, limited examinations could have a significant impact on the detection of inservice degradation. In cases where examination methods are not practical or appropriate, RG 1.178 states that alternative inspection intervals, scope and methods should be developed to ensure that piping degradation is detected and structural integrity is maintained. To address this aspect, a stepped approach to limited examinations will be incorporated into WCAP-14572 that may include the examination of adjacent elements and more frequent pressure testing and visual examination for leakage. However, it should be noted that, in accordance with the regulations, limited examinations must be documented and submitted to the staff as relief requests for review and approval.

The qualification of NDE personnel, processes and equipment must comply with Section XI of the ASME Code to meet the requirements of 10 CFR 50.55a. In general, this means procedures must be qualified in accordance with ASME Section XI, Appendix VIII, or in the spirit of Appendix VIII, for techniques. As discussed in response G-19 in the NEI submittal dated March 13, 1997 (Ref. 17), Westinghouse stated that the reference plant "would qualify methods, procedures, personnel, and equipment to a level commensurate with the intent of an Appendix VIII performance demonstration."

Section 4 of WCAP-14572, "Inspection Program Requirements," notes that the use of a number of Code Cases is recommended (i.e., N-416-1, N-498-1, N-532). Staff acceptance of the WOG approach does not automatically imply acceptance of the referenced Code Cases. Licensees proposing to use the WOG approach must submit separate proposed alternatives to use these or other unapproved Code Cases.

Implementation of a RI-ISI program for piping should be initiated at the start of a plant's next ISI interval, consistent with the requirements of the ASME Code Section XI Edition and Addenda committed to by an Owner in accordance with 10 CFR 50.55a, or any delays granted by the NRC staff. In addition to other changes in Section 4.5 of WCAP-14572, Westinghouse stated in the public meeting on September 22, 1998 [item 20, Ref. 8], that the following sentence will be added in the next revision of WCAP-14572:

> "Documentation of program updates shall be kept and maintained by the Owner on site for audit. Changes arising from the program updates should be evaluated using the change mechanisms described in existing applicable regulations (e.g., 10 CFR 50.55a and Appendix B to 10 CFR Part 50) to determine if the change to the RI-ISI program should be reported to the NRC."

The staff finds the periodic reporting requirements to be acceptable since they meet the existing applicable regulations. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

WCAP-14572, Revision 1 states that periodic updates of RI-ISI programs will be performed at least on a period basis to coincide with the inspection program requirements contained in ASME Section XI under Inspection Program B. The staff finds these updates acceptable because they meet ASME Section XI which requires updates following the completion of all scheduled

examinations in each inspection interval. WCAP-14572 also states that RI-ISI programs will be evaluated for changes in safety-significance and inspection requirements due to plant design feature changes, plant procedure changes, equipment performance changes, and examination results including flaws or indications of leaks. This process for RI-ISI program updates meets the guidelines of RG 1.174 that risk-informed applications must include performance monitoring and feedback provisions and hence is acceptable to the staff.

3.6 Conformance to Regulatory Guide 1.174

RG 1.174 describes an acceptable method for assessing the nature and impact of licensing basis changes by a licensee when the licensee chooses to support these changes with risk information. This Reg Guide identifies a four-element approach for evaluating such changes, and these four elements are aimed at addressing the five principles of risk-informed regulation. Section 1.4 of WCAP-14572 Revision 1 summarizes how the proposed WOG RI-ISI process conforms to the RG 1.174 approach. The staff finds that WCAP-14572 approach is consistent with RG 1.174 as discussed below.

In Element 1 of the RG 1.174 approach, the licensee is to define the proposed change. Section 1.1 of WCAP-14572 discusses current regulatory requirements for the ISI program and the changes in regulatory compliance using the RI-ISI approach. The scope of the changes is also discussed, and this scope includes the addition of non-ASME code piping that has been identified as high safety significant. The staff finds that the discussion in Section 1.1 of WCAP-14572 to be consistent with the guidance provided in Section 2.1 of RG 1.174.

Element 2 is the performance of the engineering analysis. In this element, the licensee is to consider the appropriateness of qualitative and quantitative analyses, as well as analyses using traditional engineering approaches and those techniques associated with the use of PRA findings. Regardless of the analysis method chosen, the licensee must show that the principles set forth in Section 2 of RG 1.174 have been met. The staff finds that the evaluation process as described in Section 3 of WCAP-14572 meets the requirements of this Element. WCAP Section 3 describes the probabilistic and deterministic engineering analyses to be performed and integrated through the use of a plant expert panel to define the high and low safety significant piping segments. The results of these analyses are used to select the inspection locations and inspection methods, and a statistical model is used to determine the number of locations to be inspected to meet confidence and reliability goals.

Element 3 is the definition of the implementation and monitoring program. The primary goal of this element is to ensure that no adverse safety degradation occurs because of changes to the ISI program, and the staff finds that the guidance provided in WCAP Section 4.5 is adequate to meet this goal. Section 4.5 of WCAP-14572 discusses how the implementation of the RI-ISI program is consistent with the requirements of ASME Code Section XI. In addition, the monitoring, feedback and corrective action program discussed is consistent with guidelines provided in Section 2.3 of RG 1.174.

Element 4 is the submittal of the proposed change. WCAP-14572 states that each licensee will submit their proposed change at the time they perform a RI-ISI program.

RG 1.174 states that, in implementing risk-informed decision-making, plant changes are expected to meet a set of key principles. The paragraphs below summarize these principles, and staff findings with regard to the conformance of WCAP-14572 methodology with these principles.

Principle 1 states that the proposed change must meet current regulations unless it is explicitly related to a requested exemption or rule change. The proposed RI-ISI change is an alternative to the ASME Section XI Code as referenced by 10 CFR 50.55a(a)(3) for piping ISI requirements with regard to the number of inspections, locations of inspections, and methods of inspections.

Principle 2 states that the proposed change must be consistent with the defense-in-depth philosophy. ISI is an integral part of defense-in-depth. It is expected that as part of the RI-ISI process, the safety significance categorization, the expert panel review and approval, and the subsequent number and location of elements to inspect will maintain the basic intent of ISI (i.e., identifying and repairing flaws before pipe integrity is challenged). Therefore, although a reduction in the number of welds inspected is anticipated, it is expected that there will be reasonable assurance that the program will provide a substantive ongoing assessment of piping condition.

Principle 3 states that the proposed change shall maintain sufficient safety margins. No changes to the evaluation of design basis accidents in the final safety analysis report (FSAR) are being made by the RI-ISI process. In addition, Section 3.7 of WCAP-14572 describes the use of a statistical model to assure that safety margins (in terms of pipe failure probability) are maintained. This statistical model is based on the evaluation of potential flaws and leakage rates that are precursors to piping failure.

Principle 4 states that, when proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement. Sections 1.4, 3.6, 3.7, and 4.4 of WCAP-14572 provide arguments that a RI-ISI program is, as a minimum, a risk-neutral application and should result in a risk reduction. Staff findings with regard to principle 4 are found in Section 3.3.3 of this SER.

Principle 5 states that the impact of the proposed change should be monitored using performance measurement strategies. WCAP-14572 conformance to this principle is already discussed in the paragraph on Element 3 above.

4.0 CONCLUSIONS

10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The staff concludes that the proposed RI-ISI program as described in WCAP-14572, Revision 1, conditioned upon the changes to be incorporated as discussed in Ref. 8, will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a for the proposed alternative to the piping ISI requirements with regard to the number of inspections, locations of inspections, and methods of inspections. This conclusion is founded on the findings discussed in the

remainder of this section.

The methodology conforms to the guidance provided in RGs 1.174 and 1.178, in that applying the methodology results in risk-neutrality or risk-reduction for the piping addressed in the RI-ISI program. According to this methodology, the licensees will identify those aspects of the plants' licensing bases that may be affected by the proposed change, including rules and regulations, FSAR, technical specifications, and licensing conditions. In addition, the licensees will identify all changes to commitments that may be affected as well as the particular piping systems, segments, and welds that are affected by the change in the ISI program. Specific revisions to inspection scope, schedules, locations, and techniques will also be identified, as will plant systems and functions that rely on the affected piping. The WOG procedure to subdivide piping systems into segments is founded on portions of piping having the same consequences of failure to be placed into the same piping segments. In addition, consideration is given to identifying distinct segment boundaries at branching points, locations of pipe size changes, isolation valve, and MOV and air-operated valves (AOV) locations.

Each segment's potential for failure is appropriately represented as failure on demand, unavailability, or frequency of failure. The relative potential for failure is consistent with systematic consideration of degradation mechanisms, segment and weld material characteristics, and environmental and operating stresses. The assessment of component failure potential attributable to aging and degradation takes into account uncertainties. Computer codes used to generate quantitative failure estimates have been verified and validated against established industry codes. Supplement 1 to WCAP-14572, Revision 1, describes the models, software, and validation of the SRRA computer code. The SRRA model is used to estimate the probability of piping failures. Peer reviews of the SRRA code have been performed on several occasions. The author of the code has published several papers for presentation at technical conferences, with technical peer reviews being part of the publication process. Earlier versions of the code have been used by Westinghouse in past research projects which have also been reviewed by the staff. In addition, the methodology of the code parallels approaches used in other generally accepted probabilistic structural mechanics codes, such as pc-PRAISE. Technical reviews of the SRRA code were performed during the Surry Unit 1 pilot plant study by the staff, its contractors, and the ASME Research Task Force on Risk-Based Inservice Inspection. These efforts provided a detailed review of the Westinghouse SRRA code, and comments from this effort resulted in several improvements to the SRRA code, as reflected in WCAP-14572, Revision 1, Supplement 1. The recent reviews were based on (1) documentation of the code, (2) detailed descriptions of example calculations, (3) trial calculations performed with the SRRA code by peer reviewers, and (4) benchmark calculations to compare failure probabilities predicted by the SRRA code and the pc-PRAISE code.

The stress corrosion cracking model of the SRRA code has a relatively simple technical basis, which does not attempt to model the complex failure mechanism in a detailed mechanistic manner. The calculations are based on a number of significant assumptions as discussed in Section A.4.3 of this SER. In particular, the code documentation given in WCAP-14572, Revision 1, Supplement 1, acknowledges the limitations of the model, and recommends the use of the pc-PRAISE computer code if predictions from a more refined mechanistic model are needed. The probabilistic fracture mechanics calculations for IGSCC have not been benchmarked for consistency with plant-specific and industry operating experience. In this regard, the Surry Unit 1 evaluations do not provide a particularly good basis to evaluate the

SRRA stress corrosion cracking model, because IGSCC makes only a small contribution to piping failures for PWR plants. The staff therefore requires that the IGSCC model be further evaluated on future applications to BWR plants, because IGSCC is a major factor governing piping integrity at BWRs.

The staff noted several limitations, e.g., IGSCC modeling, lack of benchmarking of E-C model compared to existing E-C programs, lack of modeling of complex geometries, etc. in the SRRA code. These limitations in the SRRA code result in a need for judicious use of the code and careful attention by the expert panel to ensure that the results of the code seem appropriate. It should be noted that the use of SRRA, or other probabilistic fracture mechanics codes, to estimate relative failure frequencies of piping systems and components is appropriate, but that the ability of such codes to estimate failure frequencies is limited by the quality of the input data and modeling limitations inherent in the code itself. Providing bounding or conservative inputs to the model or relying on the conservative nature of certain aspects of the code can potentially lead to inappropriate conclusions regarding the relative susceptibility to failure of various piping segments and components. Therefore, it is extremely important that these limitations be recognized by the user of the code and by the licensees' expert panel and that the results of the analyses are carefully scrutinized to assure that they make sense when compared to engineering knowledge of degradation mechanisms and plant specific and generic operating experience. Further details of the limitations and staff recommendations on the use of the SRRA code are provided in Section A.25 of this SER.

The impact on risk attributable to piping pressure boundary failure considers both direct and indirect effects. Consideration of direct effects includes failures that cause initiating events or disable single or multiple components, trains or systems, or a combination of these effects. The methodology also considers indirect effects of pressure boundary failures affecting other systems, components and/or piping segments, also referred to as spatial effects such as pipe whip, jet impingement, flooding or failure of fire protection systems.

The results of the different elements of the engineering analysis are considered in an integrated decision-making process. The impact of the proposed change in the ISI program is founded on the adequacy of the engineering analysis, acceptable change in plant risk, and the adequacy of the proposed implementation and performance monitoring plan, in accordance with RG 1.174 guidelines.

WOG methodology also considers implementation and performance-monitoring strategies. Inspection strategies ensure that failure mechanisms of concern have been addressed and there is adequate assurance of detecting damage before structural integrity is impacted. Safety significance of piping segments is taken into account in defining the inspection scope for the RI-ISI program.

System pressure tests and visual examination of piping structural elements will continue to be performed on all Class 1, 2, and 3 systems in accordance with the ASME BPVC Section XI program, regardless of whether the segments contain locations that have been classified as HSS or LSS. The RI-ISI program applies the same performance measurement strategies as existing ASME requirements and, in addition, broadens the inspection volumes at weld locations.

WCAP-14572, Revision 1, has provided the methodology to conduct an engineering analysis of the proposed changes using a combination of engineering analysis with supporting insights from a PRA. Defense-in-depth and quality is not degraded in that the methodology provides reasonable confidence that any reduction in existing inspections will not lead to degraded piping performance when compared to existing performance levels. Inspections are focused at locations with active degradation mechanisms as well as selected locations that monitor the performance of the front-line primary system piping (the second barrier of fission product release).

Safety margins used in design calculations are not changed. Piping material integrity is monitored to ensure that aging and environmental influences do not significantly degrade the piping to unacceptable levels.

Augmented examination program for degradation mechanisms such as IGSCC and EC would remain unaffected by the RI-ISI program and WCAP-14572 should not be taken as a basis to change the augmented inspection program.

Although the staff finds that the general guidance provided in WCAP-14572 Revision 1 (and as amended by Ref. 8) to be acceptable, application of this guidance will be plant-specific. As such, individual applications in RI-ISI must address the various plant-specific issues. These include:

- The quality, scope and level of detail of the PRA used, as described in RG 1.174 and 1.178 (see Section 3.3 and 3.3.1 of this SER).

- The guidelines and assumptions used for the determination of direct and indirect effects of flooding, including assumptions on the failure of components affected by the pipe break (see Sections 3.2.4 and 3.3.1 of this SER)

- The criteria, and the justification for the criteria used for the categorization of piping segments, including sensitivity studies to model human actions and segment failure probability (see Section 3.3.2 of this SER).

In the public meeting on October 8, 1998 (Ref. 18), the staff and the industry discussed the information to be submitted to the NRC and the list of retrievable onsite documentation for potential NRC audits of licensees that seek to utilize the WOG methodology for their RI-ISI program. The staff's expectation is that contents of submittals to NRC listed below will consist of brief statements and results of program development with details available as retrievable onsite documentation for potential NRC audits:

- Submittal Contents

(1) justification for statement that PRA is of sufficient quality
(2) summary of risk impact
(3) current Inspection Code
(4) impact on previous relief requests
(5) revised FSAR pages impacted by the change, if any
(6) process followed (WCAP, Code Case, and exceptions to methodology, if any)

(7) summary of results of each step (e.g., number of segments, number of HSS and LSS segments, number of locations to be inspected, etc.)
(8) a statement that RG principles are met (or any exceptions)
(9) summary of changes from current ISI program
(10) summary of any augmented inspections that would be impacted

- Retrievable Onsite Documentation for Potential NRC Audit

(1) scope definition
(2) segment definition
(3) failure probability assessment
(4) consequence evaluation
(5) PRA model runs for the RI-ISI program
(6) risk evaluation
(7) structural element/NDE selection
(8) change in risk calculation
(9) PRA quality review
(10) continual assessment forms as program changes in response to inspection results
(11) documentation required by ASME Code (including inspection personnel qualification, inspection results, and flaw evaluations)

5.0 REFERENCES

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2. Westinghouse Topical Report, WCAP-14572, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment (SRRA) Model for Piping Risk-Informed Inservice Inspection," Revision 1, October 1997.

3. U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.

4. U.S. Nuclear Regulatory Commission, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping," Regulatory Guide 1.178 for Trial Use, September 1998.

5. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Code Case N-577, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method A," September 2, 1997.

6. Standard Review Plan Chapter 3.9.8, "Standard Review Plan for Trial Use for the Review of Risk-Informed Inservice Inspection of Piping," October 1998.

7. Standard Review Plan Chapter 19.0, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," NUREG-0800, July 1998.

8. Letter from Louis F. Liberatori, Jr., Chairman (Westinghouse Owners Group), to Peter C. Wen (NRC), "Transmittal of Responses to NRC Open Items on WOG RI-ISI Program and Reports WCAP-14572, Revision 1, and WCAP-14572, Revision 1, Supplement 1," September 30, 1998.

9. Letter from Peter C. Wen (NRC), to Andrew Drake (Westinghouse Owners Group), "Open Items Related to Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection (WCAP-14572, Revision 1)," September 2, 1998.

10. Letter from P. Krishnaswamy (Battelle-Columbus), to Deborah Jackson (NRC), "Independent Peer Review of Draft Regulatory Guide DG-1063," October 31, 1997.

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13. American Nuclear Society, "Design Basis for Protection of Light-Water Nuclear Power

Plants Against the Effects of Postulated Pipe Rupture," ANSI/ANS-58.2-1988, 1988.

14. Westinghouse Systems Standard Design Criteria (SSDC) 1.19, "Criteria for Protection Against Dynamic Effects Resulting from Pipe Rupture," Revision 1, 1980.

15. U.S. Nuclear Regulatory Commission NUREG-1661," Technical Elements of Risk-Informed Inservice Inspection Programs for Piping" to be published.

16. Letter from Harry Martz (Los Alamos National Laboratory), to Lee Abramson (NRC), "Independent Peer Review of Draft Regulatory Guide DG-1063," September 5, 1997.

17. Letter from Anthony R. Pietrangelo (Nuclear Energy Institute), to Dr. Brian W. Sheron (NRC), containing responses to NRC's Request for Additional Information, March 13, 1997,

18. Minutes for NRC Meeting with Nuclear Energy Institute (NEI) Regarding Risk-Informed Inservice Inspection Programs on October 8, 1998.

APPENDIX A

Review of WCAP-14572, Revision 1, Supplement 1, "Westinghouse Structural Reliability and Risk Assessment Model for Piping Risk-Informed Inservice Inspection"

A.1 INTRODUCTION

Supplement 1 to WCAP-14572, Revision 1, describes the models, software and validation of the SRRA computer code. The SRRA model is used to estimate the probabilities of piping failures, which are input to the PRA in support of the WOG RI-ISI program for piping.

A.2 Background

RG 1.178 provides an option for licensees to quantitatively estimate the reliability of individual pipe segments within the scope of the RI-ISI program. These estimates are to be consistent with industry databases on piping failure rates and relevant to plant-specific operating experiences. Detailed knowledge of piping design parameters, materials degradation mechanisms, plant operating conditions, and the likelihood of fabrication and service-induced flaws are elements of a quantitative analysis that need consideration. The use of probabilistic structural mechanics computer codes is an acceptable approach to estimate structural failure probabilities on the basis of such detailed knowledge.

The SRRA computer software was developed by the Westinghouse Electric Company over the last decade and has been enhanced to support the development of risk-informed inservice inspection programs of piping. This software was applied in plant applications of the RI-ISI program development for the Millstone Unit 3 and Surry Unit 1 nuclear power plants. The NRC staff and contractor personnel were briefed at public meetings during the course of these pilot applications. During these studies and methods development activities, the SRRA code was enhanced as issues were identified and resolved.

The current review was performed recognizing that probabilistic structural mechanics codes, including the SRRA code, are limited in their ability to predict absolute values of failure probabilities with a high degree of accuracy. The models themselves, along with the various inputs needed to apply these models, are subject to many uncertainties. In addressing the value of a given computer code to calculate failure probabilities the following considerations were taken to be important:

- While it is expected that advances in the technology will someday reduce the levels of uncertainty in calculated failure probabilities, the ability of the models to estimate relative failure probabilities is considered to be more important than their ability to predict absolute values. In this regard, RI-ISI is largely governed by relative values if risk both for the ranking and selection of components to be inspected and for the evaluation of risk increases or decreases associated with changes in the inspection programs.

- Relative values of failure probabilities are not used directly in the RI-ISI process. However, it is the relative values of failure probabilities along with relative values of failure consequences that are important to the final results of the risk-informed evaluations.

- It is important to the RI-ISI process to calculate absolute values of failure probabilities as accurately as possible, because an increased levels accuracy and consistency in the calculations will contribute to a corresponding enhancement in the accuracy of the relative values of failure probabilities.

- The calculation of failure probabilities with codes such as SRRA should not be performed in isolation of other independent methods of estimating failure probabilities, such as data bases and plant operating experience. Results of calculations should always be evaluated for reasonableness and consistency, and the assumptions and inputs to the calculations should be refined as appropriate.

A.3 Overview of Assessment

Over the past 3 years, as ASME-Research and WOG developed methods to perform RI-ISI of piping, the staff held public meetings with both groups to develop guidelines for acceptable uses of probabilistic fracture mechanics computer codes. In addition, with the assistance of Pacific Northwest National Laboratory (PNNL), the staff performed independent audit calculations to validate the results of the SRRA computer code.

The following discussion addresses the strengths and limitations of the Westinghouse SRRA computer code. Given the broad scope of piping designs and operating conditions, it was not expected that any one computer code could address all of the failure mechanisms and piping designs encountered in a nuclear power plant. Therefore, a key part of this review focused on the documentation for the Westinghouse code and how well it achieved the following objectives:

(1) Inform the code user about code limitations.
(2) Provide technically sound guidance on alternative approaches to estimate piping failure probabilities.

Important elements of this evaluation include the equations and assumptions (inputs) used in the piping reliability models, as well the validation of the estimated failure probabilities. In some cases, it is appropriate to place certain detailed inputs outside the direct control of the user (incorporating inputs into the model itself). In other cases, specific recommendations can be provided in the user document with example problems. Where possible, input values were standardized for specific applications. Many of these inputs were the subject of significant discussions during periodic public meetings on the Surry Unit 1 pilot applications, and are addressed in this review.

A.4 REVIEW OF SPECIFIC ISSUES

This section addresses specific aspects of the probabilistic structural mechanics model from the standpoint of the consistency and reasonableness of the estimated failure probabilities.

A.4.1 Failure Mechanisms

As described in the following sections, the Westinghouse SRRA code addresses with various levels of detailed modeling the degradation mechanisms of (1) fatigue, (2) stress corrosion cracking, and (3) flow-assisted corrosion/wastage or wall thinning. The present review concludes that acceptable technical approaches are used for each of these mechanisms.

A.4.2 Fatigue

The fatigue model assumes that all failures by this mechanism result from preexisting flaws. Inputs to the model are sufficiently flexible to address low cycle fatigue attributable to normal plant transients, high cycle fatigue from thermal fatigue (resulting, for example, from stratification of fluids), and high cycle vibrational fatigue.

Calculations are based on a relatively detailed mechanistic model which relates fatigue crack growth to the amplitude and frequency of the cyclic stresses. The Westinghouse/SRRA model for fatigue is very similar to that used in the NRC developed pc-PRAISE code, and numerical results of the SRRA code have been successfully benchmarked (as described later) against results from the pc-PRAISE code.

In common with the pc-PRAISE code, Supplement 1 to WCAP-14572 does not address fatigue crack initiation except in an indirect manner by conservatively assuming that initiated cracks are present at the beginning of plant operation. The limitations of this approach to fatigue crack initiation are addressed below.

In common with the pc-PRAISE code, fatigue cracks are all conservatively assumed to be located at the pipe inner surface. Crack growth in both the depth direction (through-wall direction) and in the length direction are simulated in a manner essentially the same as that used in the pc-PRAISE code.

The SRRA code permits the simulation of uncertainties in the levels of low and high fatigue stress cycles, which treats the amplitude of fatigue stress as a deterministic parameter.

The staff concludes that the SRRA code addresses fatigue crack growth in an acceptable manner since it is consistent with the technical approach used by other state-of-the-art codes for probabilistic fracture mechanics. It should be noted, however, that realistic predictions of failure probabilities require that the user define input parameters, which accurately represent all sources of fatigue stress and the probabilities for preexisting fabrication cracks in welds. The major limitation of the model is its inability to realistically simulate the initiation of fatigue cracks, which experience has shown to be the primary contributor to fatigue failures at operating plants.

A.4.3 Stress Corrosion Cracking

The stress corrosion cracking model of the SRRA code has a relatively simple technical basis, which does not attempt to model the complex failure mechanism in a detailed mechanistic manner. The calculations are based on a number of significant assumptions as follows:

- All piping failures by this mechanism result from preexisting fabrication flaws, although service experience with stress corrosion cracking indicates that such failures are dominated by cracks in welds that initiate during plant operation.

- The effects of crack initiation can conservatively be estimated by assuming one flaw per weld at the start of plant operation, with the flaw size distribution being the same as that for

welding-related fabrication flaws. Although calculations based on this assumption can provide relative probabilities of failure for different pipe segments, it is important for the expert panel to review the predicted failure probabilities to ensure a selection of input parameters that provides predictions, which are reasonable and consistent with plant operating experience.

- There is sufficient knowledge on the part of the plant technical staff and the expert panel (in combination with plant operating history with the occurrence of IGSCC) of the plant-specific environmental factors (water chemistry, temperature, etc.), levels of weld sensitization, and residual stress levels to identify pipe segments that have a high, medium or low potential for failure by stress corrosion cracking.

- The probability of through-wall cracks for the high failure potential case can be calculated using a bounding crack growth rate curve developed in 1988 (NUREG-0313), this curve relates crack growth rates to crack tip stress intensity factors.

- IGSCC related crack growth rates of moderate and none are assigned in the SRRA code to be a factor of 0.5 and 0.0 less than the bounding rate, with engineering judgement used to assign crack growth rates to these broad categories. Alternatively, the SRRA user can directly assign a numerical factor to be applied to the bounding crack growth rates.

In summary, the stress corrosion cracking model of the SRRA code provides a systematic basis to translate inputs into estimated failure probabilities on the basis of engineering judgement and operating experience. The model combines the inputs for stress corrosion cracking with other factors such as pipe dimensions and applied loads to predict pipe failure probabilities. While some of the modeling assumptions appear to be quite conservative, the calculations for the Surry Unit 1 plant appear to predict reasonable trends.

In particular, the code documentation given in WCAP-14572, Revision 1, Supplement 1, acknowledges the limitations of the model, and recommends the use of the pc-PRAISE computer code if predictions from a more refined mechanistic model are needed. The probabilistic fracture mechanics calculations for IGSCC have not been benchmarked for consistency with plant-specific and industry operating experience. In this regard, the Surry Unit 1 evaluations do not provide a particularly good basis to evaluate the SRRA stress corrosion cracking model, because IGSCC makes only a small contribution to piping failures for PWR plants. The staff therefore requires that the IGSCC model be further evaluated on future applications to BWR plants, because IGSCC is a major factor governing piping integrity at BWRs.

A.4.4 Flow Assisted Corrosion/Wastage

The wastage model of the SRRA code has a relatively simple technical basis and does not attempt to model the complex wall thinning processes in a detailed mechanistic manner. Deterministic models, such as the CHECKWORKS code developed by the Electric Power Research Institute (EPRI) are available to relate wall thinning rates to basic parameters such as flow velocity, chemical composition of the pipe material, fluid temperature, single-phase water versus two-phase steam/water mixture, and pH level of the fluid. However, probabilistic forms of

such deterministic models have not yet been developed.

While a close reading of the code documentation as given in WCAP-14572, Revision 1, Supplement 1, provides information on assumptions made in the SRRA wall thinning model, many users could have difficulty relating inputs to the model to the type of information available to plant technical staff. In addition, users may not have sufficient insight into the assumptions behind the wall thinning model to perform calculations in a correct and consistent manner. However, the calculations for Surry Unit 1 had sufficient participation by the Westinghouse staff to ensure that calculations for the Surry Unit 1 study yielded reasonable results.

Supplement 1 to WCAP-14572, Revision 1, provides information on assumptions made in the SRRA wall thinning model. Before issuing of this SER, the staff expressed a concern that many users could have difficulty relating inputs to the model with the type of information available to plant technical staff. In addition, users may not have sufficient insight into the assumptions behind the wall thinning model to perform calculations in a correct and consistent manner. Consequently, the staff indicated that WCAP-14572 should provide guidance for plant personnel executing the SRRA code for flow-assisted corrosion (FAC) that provides reasonable assurance that the results calculated for FAC failure probabilities are appropriate. In the public meeting on September 22, 1998 [item 7 (b), Ref. 8], Westinghouse stated that the next Revision of WCAP-14572 will provide guidance for material wastage potential. The staff concludes that the guidance for estimating the material wastage potential is acceptable since, if material wastage rates are high enough to proceed through the pipe wall, the probabilities of small leak, large leak and break are all calculated to be the same. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

The wall thinning model in the SRRA code is based on the following assumptions:

- The user of the code is able to estimate the rate of wall thinning (e.g., inches of wall thickness reduction per year) and express this rate in terms of a "best estimate" value and a distribution function (e.g., log-normal distribution) that describes the variability or uncertainty associated with the best estimate.

- Wall thinning can be treated in a simplified manner by assuming that the maximum local rate of thinning occurs uniformly over a substantial length of straight pipe; this is a conservative assumption which does not account for variations (reduced rates of thinning) in the axial or circumferential directions as is case for the important case of local wall thinning at elbow locations.

- Consistent with the previous assumption, all failures of piping resulting from wall thinning will result in pipe breaks rather than leakages; pipe failures will occur when the simulated level of pressure-induced hoop stress becomes equal to the simulated values of the flow stress of the piping material.

Data from industry experience, along with structural mechanics considerations of localized thinning, provide evidence that leak-before-break events are more likely than sudden pipe breaks. The assumption that leak-before-break does not apply, as used in the SRRA code, is a conservative assumption.

The input parameter for the wall thinning rate is expressed in a simplified manner in the SRRA code with a parameter of 1.0 being assigned whenever the user believes that the thinning rate is high. The code assigns a "best estimate" thinning rate of 0.0095 inch per year for this rate parameter along with a variability described by a log-normal distribution which implies that the natural logarithm of the thinning rate has a standard deviation of 0.893 (which corresponds to a value of 2.3714 for the so called "deviation or factor" used as input to the SRRA code). For a rate parameter other than 1.0, the best estimate of the thinning rate is assigned to be proportional to the selected value of the parameter.

The staff concludes that plant technical personnel have sufficient knowledge and field measurements of wall thinning rates to develop reasonable inputs to the SRRA code for estimating failure probabilities for FAC degradation mechanisms. Such information is generally available as a result of the ongoing programs for flow-assisted corrosion which are required at all plants. The approach uses data and/or engineering judgement to estimate a wall thinning rate. The probabilistic structural mechanics model then calculates failure probabilities based on the estimated thinning rates, in combination with other governing parameters such as the pipe dimensions, applied stresses, and material strengths.

Calculations with the model must be closely coordinated with the existing plant programs for the management of wall thinning, because the model requires inputs that can be obtained only from the knowledge gained from ongoing monitoring and evaluations of wall thinning rates. Furthermore, application of the probabilistic model of the SRRA code should not be used to make changes in existing programs for the inspection and monitoring of piping for wall thinning.

A.4.5 Failure Modes (Leaks and Breaks)

The staff finds the code's failure modes capabilities acceptable for RI-ISI application since the SRRA code was modified during the Surry Unit 1 pilot application to address the failure mode of large system-disabling leaks in addition to the failure modes of small leaks (through-wall cracks) and pipe breaks. The disabling leak rate for each system is assigned to be consistent with existing evaluation of plant operational and safety evaluations. The modified program can address the various modes of pipe failure corresponding to consequences identified in plant PRAs and safety analysis reports.

A.5 Component Geometries

The SRRA code was developed to address the simple geometry of a circumferential flaw in a girth welded pipe joint. In this regard, the SRRA code has a capability similar to that of other state-of-the-art probabilistic fracture mechanics codes such as pc-PRAISE.

Application of SRRA to other more complex component geometries (e.g., elbow and tee pipe fittings) requires conservative assumptions founded on treating the maximum local stresses as uniform through the pipe wall, with no credit taken for the mitigating effects of stress gradients. Calculations by Khaleel and Simonen (1997) have shown that this assumption can result in failure probabilities being overestimated by an order of magnitude or more.

With proper attention to stress inputs and the interpretation of calculated results, the SRRA code can be used effectively to estimate failure probabilities for components with more complex geometries. Before issuing this SER, the staff identified an open item that WCAP should provide guidance for the analyst on the code limitations for complex geometries and guidance for effective use of the code in such applications. In the public meeting on September 22, 1998 [item 12, Ref. 8], Westinghouse stated that the SRRA piping models only apply to standard piping geometry (circular cylinders with uniform wall thickness). Westinghouse further stated that a limitation on the use of non-standard geometry will be added in the next revision of WCAP-14572. The staff finds this clarification of the code limitation to be acceptable. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

A.6 Structural Materials

For calculational convenience, structural reliability computer codes should be able to address a range of piping materials. The capabilities of the SRRA code meets this criterion. The code has generally been applied in a mode which uses simplified inputs consistent with standardized material properties for stainless and ferritic piping materials. However, the code can also be operated in a mode which allows greater flexibility for the specification of input parameters for material properties. The staff recommends that licensees apply the code in a manner that accounts for the known plant-specific material characteristics as they may be governed by such factors as carbon content, heat treatments, etc.

As with any computer code, the quality of results often depends on the capabilities of the code user. In this case, the user must first recognize situations for which it is inappropriate to use the standard menu selections of material properties. Before issuing this SER, the staff indicated that WCAP-14572 should specify the level of training and qualification that the code user needs to properly execute the SRRA code. In its response in the public meeting on September 22, 1998 [item 13, Ref. 8], Westinghouse indicated that the next revision of WCAP-14572 will state that to ensure that the simplified SRRA input parameters are consistently assigned and the SRRA computer code is properly executed, the engineering team for SRRA input should be trained and qualified. The revised WCAP will also list the topics covered in this training as presented in the public meeting on September 22, 1998 [item 13, Ref. 8]. The staff has reviewed the additional guidance for training and qualification and determined that it provides reasonable assurance that code users will be able to properly execute the SRRA code. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

A.7 Loads and Stresses

The SRRA code has several inputs to describe the loads and stresses that govern piping failure. The stresses used for plant specific applications should be based on actual plant experience and operational practices (including thermal and vibrational fatigue stresses), which may differ from the stresses used for purposes of the original design of the plant. The types of stresses of concern include residual and vibrational (fast transient) stresses which are specifically addressed below. Other inputs address low cycle fatigue (slow transients) and design-limiting stresses which include the effects of seismic loadings. For applications of RI-ISI programs to

actual plants, plant-specific inputs such for loads and stresses should be used.

All calculations assume that the stresses are uniformly distributed through the thickness of the pipe wall. This simplifying assumption is conservative and could be avoided (with methods currently used in the pc-PRAISE code).

The inputs for low cycle fatigue can address only one type of loading transient, which is assumed to represent the dominant contribution to fatigue crack growth, although well-known methods exist to evaluate the combined effects of many operational transients. However, limiting the evaluation to one dominant transient is a reasonable approach, given the intended scope of the SRRA code, which is to estimate failure probabilities using simplified approaches.

Similarly, the SRRA code requires the user to select a single level of design-limiting stresses and an associated occurrence frequency which best characterizes the loads governing the probabilities of a pipe break. The selection is based on plant experience, records of transients, engineering judgement or other considerations. In some cases, the normal operating loads will be more important (because they occur with a probability of 100 percent) than much larger seismic loads that have lower occurrence rates (e.g., a frequency 10^3 per year). Applications of the SRRA code before the 1996 benchmarking activity were founded on design-limiting stresses related to seismic loads, and with a standardized occurrence frequency of 10^3 per year. Discussions during the 1996 benchmarking effort noted that higher probability loads should also be addressed. These discussions led Westinghouse to use as inputs the design-limiting (e.g., pressure, dead weight, etc.) loads in combination with an occurrence frequency of once per year, or probabilistically distributed as a function of time in the calculations, an approach which may result in conservative predictions of pipe break frequencies.

The staff finds the treatment of loads and stresses as discussed above to be conservative and acceptable for the purpose of RI-ISI program application since the use of less conservative loads and stresses would require more detailed structural analyses and in most cases should not impact either the categorization process or the change in risk calculations. In reviewing plant specific calculations performed with the SRRA code it has been noted that sensitivity calculations have been used to evaluate the effects of conservative inputs for piping stress. For example, failure probabilities associated with high stresses due to postulated snubber lockup have been adjusted to account for the probability that the lockup condition will actually occur. Such evaluations are an important step to ensure that conservative inputs do not unrealistically impact the categorization and selection of piping locations to be inspected. In summary, while an appropriate selection for input parameters for loadings is a critical step in the evaluation, licensees have the needed expertise to identify the required input to the SRRA input menu.

A.8 Vibrational Stresses

The NRC staff and the industry have recommendations that address appropriate levels (as a function of pipe size) for vibrational stresses to be used in failure probability calculations. These recommendations arose from concerns regarding assumptions made for early calculations performed for Surry Unit 1 by Westinghouse and Virginia Power, and were developed with guidance from the ASME Research Task Force on Risk-Based Inspection Guidelines.

Since the Westinghouse SRRA code has incorporated the recommendations of the ASME Task Force as default values for those piping locations at which high levels of vibrational stresses are expected, the staff concludes that the treatment of vibrational stress as in the SRRA code is acceptable. The recommended levels of vibrational stresses will be fully documented in a revision to WCAP-14572. The actual piping locations where vibrational stresses are to be expected are assigned by plant technical staff on the basis of judgement taking into account such factors as proximity to rotating equipment and knowledge of plant operating experience. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

A.9 Residual Stresses

The Westinghouse SRRA code includes inputs for residual stress which describe both median values and variability in the level of stress. The residual stress contribution is an important contribution to the growth of stress corrosion cracks, and can also influence the growth of fatigue cracks through the so-called R-Ratio effect.

Appropriate levels of welding residual stress were discussed in review meetings held during the Surry Unit 1 pilot application, and a consensus was developed to guide the selection of residual stress inputs. Since the SRRA code uses the resulting recommendations which specify a log-normal distribution to describe the uncertainty in residual stress, with an upper bound on the distribution (or truncation) at 90 percent of the flow stress (corresponding to the 90th percentile of the log-normal distribution), the staff finds the treatment of residual stresses acceptable.

A.10 Treatment of Conservatism

RG-1.174 recommends that all calculations used in the categorizing risk (including the calculations of component failure probabilities) should be performed on a "best estimate" basis rather than conservatively. Conservative assumptions can introduce undesirable biases into the ranking process by masking the significance of those components for which realistic rather than conservative evaluations are performed. In the case of inservice inspections, the result could, for example, lead to an inappropriate amount of inspection of small versus large pipes, or excess inspection for stress corrosion cracking versus inspection for flow-assisted corrosion.

With a few exceptions, the Westinghouse SRRA code performs "best estimate" calculations. On the basis of this review, the staff concludes that conservative assumptions are consistent with practices used in similar computer codes, and/or are consistent with limitations of current technology to predict structural failures. Nevertheless, particular applications of the code may address uncertainties regarding code inputs by assigning very conservative values, and thereby generate inappropriately conservative estimates of failure probabilities. The present review also addresses the following potential sources of conservatism on the basis of practices used in the Surry Unit 1 pilot study:

- Inputs for the number and sizes of fabrication flaws are a significant source of uncertainty. In estimating the number of flaw in a weld, the SRRA code accounts for the volume of metal in the weld by relating this volume to the circumference and wall thickness of the pipe. The

SRRA code, like the pc-PRAISE code, places all flaws at the pipe inner surface, and in this step makes conservative assumptions about the fraction of the flaws in each given weld which should be counted as surface flaws. This estimated fraction is believed to be somewhat more conservative for thicker wall piping than for thinner wall piping, and may therefore bias inspections to larger piping.

- The treatment of stress corrosion cracking could give very conservative predictions of failure probabilities because of conservative assumptions in the structural mechanics model. In particular, the model makes three conservative assumptions:

  (1) There is a 100 percent probability that an IGSCC crack will initiate in each weld.
  (2) The crack initiates at time equals zero.
  (3) The size distribution of the initiated cracks is the same as for welding related flaws.

  Evidently, there are offsetting factors which lower the calculated crack growth rates and thereby account for a generally good correlation of the calculated failure probabilities with service experience. The reason for the good correlation with experience is not clear. However, It appears that the SRRA calculations were performed with the intent of achieving qualitative agreement with plant operating experience. In this regard, staff recommendations encourage the use of data and operating experience to augment computer models to estimate piping failure probabilities. The WCAP does not document a formal process to use experience as a means to calibrate the SRRA calculations. Nevertheless, discussions during public meetings for reviews of the Surry Unit 1 pilot application did focus on piping locations with highest values of failure probabilities with attention to the degradation mechanisms involved and how the predictions correlated with service experience. Evidently the SRRA models have been adjusted or calibrated to ensure that the piping locations with the highest potential for IGSCC have calculated failure probabilities that are generally consistent with the experience. Having "anchored" the highest values of calculated probabilities, the model permitted probabilities for locations with lower potentials to be estimated on the basis of the relative values of calculated failure probabilities.

- The review of the Surry Unit 1 pilot study indicates conservative engineering judgements used to assign cyclic and design limiting stress. One example is that vibrational stresses are often assumed to be present (with a probability of 100 percent), where in reality the identified locations only have a potential for the occurrence of such stresses. At other locations, code limiting stress levels are assigned because results of detailed stress calculations were not available. However, review of the predicted failure probabilities calculated for the Surry pilot plant showed consistency with available industry data for the frequency of vibrational failures. As in the case of failures due to IGSCC, the results of SRRA calculations for vibrational failures were reviewed during public meetings. Inputs for vibrational stress levels were refined with an objective to predict failure probabilities that were reasonable and consistent with plant operating experience . The staff, therefore, finds the selected application of conservatism for vibrational stresses acceptable.

A.11 Numerical Methods and Importance Sampling

On the basis of this review, the staff concludes that the SRRA code calculates failure

probabilities using acceptable statistical and probabilistic methods. The Monte-Carlo method as implemented in the SRRA code is a standard approach commonly used in probabilistic structural mechanics codes including the pc-PRAISE code. Importance sampling, again a common and well-accepted approach, increases the computational efficiency of the Monte-Carlo procedure by shifting the distributions for random variables to increase the number of simulated failures. The magnitude of shift applied to the variables by the SRRA code is relatively modest and is not believed to be sufficient to cause incorrect estimates of failure probabilities.

A.12 Documentation and Peer Review

Having reviewed WCAP-14572, Revision 1, Supplement 1, the staff concludes that this document, along with other referenced technical reports and papers, provides an acceptable level of documentation for the SRRA computer code.

Peer reviews of the SRRA code have also been performed on several occasions. The author of the code has published several papers for presentation at technical conferences, with technical peer reviews being part of the publication process. Earlier versions of the code have been used by Westinghouse in past research projects which have also been reviewed by the staff. In addition, the methodology of the code parallels approaches used in other generally accepted probabilistic structural mechanics codes, such as pc-PRAISE.

During the Surry Unit 1 pilot plant study, technical reviews of the SRRA code were performed by the NRC staff, its contractors, and the ASME Research Task Force on RI-ISI. These reviews provided a detailed assessment of the Westinghouse SRRA code on the basis of (1) documentation of the code, (2) detailed descriptions of example calculations, (3) trial calculations performed with the SRRA code by peer reviewers, and (4) benchmark calculations to compare failure probabilities predicted by the SRRA code and the pc-PRAISE code. Related comments resulted in several improvements to the SRRA code, as reflected in WCAP-14572, Revision 1, Supplement 1

A.13 Validation and Benchmarking

Westinghouse has used a variety of approaches to validate the ability of structural mechanics code to predict component failure probabilities. These approaches have included comparing code predictions with plant operating experience, and comparing SRRA predictions with predictions made by other probabilistic structural mechanics codes. Results of these efforts are described in WCAP-14572, Revision 1, Supplement 1, and in a recent ASME technical paper (Bishop 1997). The results of these validation efforts are reviewed in the following subsections.

A.13.1 Benchmarking Against pc-PRAISE

As part of the Surry Unit 1 pilot application during 1996, a benchmarking activity to compare results from the Westinghouse SRRA code with the pc-PRAISE code was completed. The scope of the benchmarking calculations was limited to the failure mechanism of fatigue, because both codes address this mechanism and approach the fatigue evaluation in a similar manner.

The objective of these calculations was to start with identical specifications for input parameters, and to establish whether the two codes predict the same or similar probabilities of failure for small leaks, large leaks, and pipe rupture.

The 1996 benchmarking calculations did not address the failure mechanisms of stress corrosion cracking or wall thinning caused by flow-assisted corrosion. The pc-PRAISE code does not address the failure mechanism of wall thinning, and therefore provided no means to benchmark the predictions derived using the wall thinning model from the Westinghouse SRRA code. In addition, although both codes address stress corrosion cracking, they use significantly different technical approaches which result in very different types of input parameters. Therefore, the appropriate validation approach for this failure mechanism was to validate each code on its own merits against operating experience.

NRC staff and contractors participated in the benchmarking activity, which Westinghouse staff documented in a recent paper presented at an ASME conference (Bishop 1997). This evaluation report summarizes the benchmarking procedures and (in part) the results of that effort.

A wide range of pipe sizes, material types, cyclic stress levels and frequencies, design limiting stresses, and leak detection capabilities were addressed by the calculations. While the present review describes some difficulties and issues encountered in comparing break probabilities for stainless steel piping when leak detection was included in the calculations, the present review agrees with the overall conclusion stated by Westinghouse that the calculations did successfully benchmark the calculations for the small leak, large leak, and full break probabilities..

As stated, the benchmarking calculations of the Westinghouse SRRA code against the pc-PRAISE code were limited to the mechanism of fatigue and more specifically, fatigue-related failures of piping associated with preexisting flaws in circumferential welds. The calculations excluded failures caused by service-related cracks initiated by fatigue. However, the range of cyclic stresses and cyclic frequencies was sufficiently broad to address low cycle fatigue attributable to normal plant transients, and high cycle fatigue caused by pipe vibrations or thermal fatigue conditions.

The benchmarking effort addressed concerns over the number of Monte-Carlo trials and importance sampling implemented within the Westinghouse SRRA code. Both aspects of the numerical approach were found acceptable. Results from the audit calculations led Westinghouse to increase the default number of Monte-Carlo simulations from the original value of 5000. In addition, the review established the correctness of the importance sampling approach, which in the Westinghouse SRRA code involves a shifting of distributions for the random variable in such a direction as to obtain a larger number of simulated failures. Default values for the number of shifting were judged to be modest, and unlikely to be a source of error in calculated failure probabilities. Sensitivity calculations by Westinghouse were performed to establish the amount of shifting which would degrade the accuracy of the calculated failure probabilities, and this level far exceeded the default parameters for shifting distributions.

The benchmark calculations generally showed good agreement in calculated failure probabilities. There were no areas of significant disagreement for probabilities of either small or large leaks over the full range of input parameters, which gave a very wide range of calculated

failure probabilities.

In a few cases, limited to certain calculations involving very low break probabilities, differences in calculated break probabilities amounting to several orders of magnitude were noted between results from the two codes. Calculations with the Westinghouse SRRA code gave higher break probabilities than predicted by pc-PRAISE. The pipe break probabilities were always sufficiently small so that the pipe segments would make only negligible contributions to the core damage frequency or categorization. No significant differences were observed for cases that neglected the effects of leak detection or where the piping material was ferritic steel versus stainless steel.

The benchmarking activity was concluded before all remaining differences in calculated break probabilities were resolved. As a result, some potential sources of numerical differences were not fully explored, including details of the importance sampling procedure, and the logic used to simulate the effects of leak detection. Westinghouse has put forward revised calculations that show relatively good agreement for all break probabilities.

It should be noted that there were significant differences in calculated failure probabilities for small leaks, large leaks, and pipe breaks during the first phase of the benchmarking calculations. It became clear that the codes themselves were not the source of the differences, but rather differences in the selection of numerical values for certain input parameters, which had not been adequately specified during the initial definition of the parameters for the benchmark problems. The most critical inputs were those for flaw density and size distributions, levels of vibrational fatigue stresses, and inputs for the simulation of leak detection.

Participants in the benchmarking efforts subsequently agreed to develop improved and standardized values for the critical inputs. Using results of calculations performed by Rolls Royce and Associates, the participants developed improved inputs for flaw size distributions. Inputs for vibrational stress levels were related to pipe sizes, resulting in reduced levels of vibrational stress for the largest pipe sizes. As a final step, the SRRA code was modified to simulate the effects of leak detection using a technique consistent with the state-of-the-art methodology used by the pc-PRAISE code. These changes resulted in good agreement between the two codes.

A.13.2 Validation with Operating Experience

A number of approaches can be used to validate calculated failure probabilities for consistency with plant operation experience. The documentation given in WCAP-14572, Revision 1, Supplement 1, provides two acceptable examples of such validation for the SRRA code. Both examples address failure mechanisms (FAC and IGSCC) for which there have been a sufficient number of field failures to provide data to permit benchmarking of calculated failure probabilities with observed failure rates. The staff found acceptable the agreement between predictions and operating experience for both failure mechanisms.

For most piping segments, calculations with the SRRA code have predicted relatively small values for failure probabilities. The results indicate that failures for such pipe segments would not be expected to occur for the limited number of years of plant operation accumulated to date. The SRRA code has therefore been shown to predict very low failure probabilities for those

failure mechanisms and piping locations which have exhibited a high level of operational reliability.

The predicted failure probabilities predicted by the SRRA code for the Surry Unit 1 plant have been reviewed from the standpoint of plant-wide trends. The net plant-wide calculated failure frequency (accounting for all pipe segments and all systems) indicates about one pipe leak per year for the entire plant, and a few pipe breaks over the 40-year operating life of the plant. These predictions of overall failure rates, predicted degradation mechanisms, and the most likely locations for piping failures show an acceptable level of agreement with plant operating experience. However, as noted above, most piping locations have experienced no failures or detectable degradation, and for these locations the operating experience provides no means to validate the correctness of the relative values of calculated failure probabilities. In this regard, the RI-ISI process is designed to provide feedback of future operating experience to permit refinement of the predictive models as appropriate.

A.14 Flaw Density and Size Distributions

Inputs for the number and sizes of welding-related fabrication flaws are a large source of uncertainty in performing probabilistic structural mechanics calculations. WCAP-14572, Revision 1, Supplement 1, indicates that the SRRA code uses acceptable inputs for flaw densities and size distributions. The inputs used with the SRRA code are those developed during the 1996 benchmarking activity. These inputs were derived on the basis of trends observed in calculations generated by Rolls Royce and Associates through application of the RR-Prodigal model to simulate flaws in typical nuclear piping welds.

While there remain uncertainties in the estimated absolute values of flaw densities, the technical basis of RR-Prodigal model helps to ensure consistency in the relative values for the number and sizes of flaws as a function of pipe material, welding practice, pipe wall thickness, and volume of weld metal. The 1996 modification of the SRRA code, which included the improved means for describing flaw distributions, significantly enhanced the ability of the SRRA code to predict reasonable values (consistent with data from operating experience) for the relative failure probabilities of large diameter piping versus small diameter piping.

A.15 Initiation of Service-Induced Flaws

The fatigue and stress corrosion cracking models in the SRRA code address only failures caused by preexisting fabrication-related flaws. Such flaws are an important contribution to piping failures, particularly when the service stresses are insufficient to cause cracking of initially un-flawed material. However, many service-related failures have been associated with severe cases of cyclic stress (e.g., thermal fatigue) or aggressive operating environments (e.g., stress corrosion cracking). In these cases service-induced flaws rather than preexisting flaws are the dominant contributor to piping failures.

The documentation provided in WCAP-14572, Revision 1, Supplement 1, appropriately acknowledges the limitations of the SRRA code, and suggests that other approaches may be needed to address failures due to service-induced flaws. These methods include the pc-

PRAISE code which offers the capability to simulate the initiation of stress corrosion cracks in stainless steel welds. In this regard, the diversity of experience represented by the expert panel reviews should ensure that appropriate computer codes and data bases are used to estimate failure probabilities.

In practice, as during the Surry Unit 1 pilot study, calculations with the SRRA code have approximated service-induced flaws by assuming that one flaw per weld initiates immediately upon the start of plant operation. The size of this flaw is described by the same distribution used to describe welding-related flaws. This model is an acceptable basis to calculate conservative or bounding values of failure probabilities. However, failure probabilities calculated using this approach must be used with caution, because the overly pessimistic predictions could result in assigning inappropriately high rankings to certain pipe segments at the expense of other components which could have larger contributions to risk.

A.16 Preservice Inspection

There are no simulations within the SRRA code to account for preservice inspections as a means to reduce the number of initial fabrication flaws. Effects of preservice inspections must be included indirectly through the inputs for flaw densities and size distributions. The staff finds the flaw distribution parameters described in WCAP-14572, Revision 1, Supplement 1, to be acceptable since they were derived from predictions by the RR-Prodigal flaw simulation model, which accounts for the effects of inspections performed after completion of welding. Using these input parameters, the calculations with the SRRA code have properly addressed the effects of preservice inspections.

A.17 Leak Detection

Consistent with the objective of calculating "best estimate" rather than conservative failure probabilities, the effect of leak detection in preventing catastrophic piping failures should be included in determining the change in CDF/LERF that lead to changes in the inspection program. The Westinghouse SRRA code includes a simulation of leak detection as an enhancement to the code made during the 1996 code benchmarking activity (It should be noted that for categorizing piping segments, leak detection is not normally credited, except for the reactor coolant system where redundant leak detection capabilities exist.). It is important that inputs to the SRRA code specify realistic values of detectable leak rates. This requires an understanding of the reliability of the techniques used to detect leaks in the various plant systems of interest.

The simplified leak rate model in the Westinghouse SRRA code is based on a correlation of calculated data on leak rates obtained from a more detailed model which is part of the pc-PRAISE code. This correlation provides an acceptable basis for addressing leak detection for the specific pressure and temperature conditions for the primary coolant loop of PWR plants having fatigue type cracks. The correlation accounts for effects of crack size, pipe stress, and internal pressure, and gives approximate predictions leak rates suitable for use in leak detection models. However, the correlation can give incorrect simulations of leak detection (due to over prediction of leak rates) for systems operating at the pressures and temperatures for BWR

plants that have IGSCC cracks with morphologies differing from those of fatigue cracks.

Before issuing this SER, the staff had identified an open item that Westinghouse should address the applicability of those correlations to other plant conditions. The staff also indicated that Westinghouse should clarify whether the SRRA code can be applied to BWRs and justify the applicability of the correlations used to calculate leak rates under BWR operating conditions. In the public meeting on September 22, 1998 [item 5 (d), Ref. 8], Westinghouse stated that the existing correlations for leak rates can be used for other plant conditions beyond the RCS and that the SRRA code can be applied to BWRs; however, care must be exercised in applying this approach to BWR piping systems, particularly those subjected to IGSCC. In addition, Westinghouse indicated that WCAP-14572 will be revised to provide guidance on addressing stress corrosion cracking. The staff finds the response acceptable since most piping susceptible to stress corrosion cracking (SCC) is also subject to fatigue loading, such as normal heat up and cool down, and the leak rate correlation for fatigue type cracks was conservatively assumed for the CLVSQ Program. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

A.18 Proof Testing

The Westinghouse SRRA code does not explicitly address the potential benefits of preservice proof tests (e.g., pressurization tests) as a means to reduce piping failure probabilities. As such, the calculated failure probabilities are likely to be somewhat conservative. Components having very low failure probabilities are likely to be those most affected by proof testing (i.e., potential service failures are attributable to very deep cracks which can be discovered during proof testing).

Proof testing can be addressed indirectly by the SRRA code with a modification to the inputs for the number and sizes of initial fabrication flaws. The proof test serves to reduce the number of very large flaws.

Before issuing this SER, the staff had identified an open item that WOG should describe how proof testing is addressed in the SRRA calculations, and should clarify what impact its neglect would have on the calculated failure probabilities and categorization. In the public meeting on September 22, 1998 [item 14, Ref. 8], Westinghouse stated that the effect on the segment risk ranking and categorization would be very small and slightly conservative. Westinghouse also indicated that the next revision of WCAP-14572 will clarify that SRRA models in LEAKPROF do not take credit for eliminating large flaws, which would fail during the pre-service hydrostatic proof tests, even though this is allowed as an input option in pc-PRAISE. The staff concludes that the approach for addressing proof testing is acceptable because Westinghouse has demonstrated that the effect of proof testing on the segment risk ranking and categorization would be very small and slightly conservative. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

A.19 Inservice Inspection

The Westinghouse SRRA code can simulate the reduction in piping failures resulting from ISI.

However, the methodology described in WCAP-14572, Supplement 1, assumes no inservice inspection for purposes of establishing risk importance measures, but does credit inservice inspection in calculating the change in CDF/LERF that results in changes to the ISI program.

Inservice inspections are simulated by the SRRA code following an approach which is similar but not identical to the pc-PRAISE code. In most cases, the approach should give acceptable predictions of the effects of inspections. Nevertheless, due care must be taken to avoid overly optimistic evaluations. Before issuing this SER, the staff had identified an open item that the probability of detection curves used in calculations need to be justified for the material type, inspection method, component geometry, and degradation mechanism that apply to the structural location being addressed. In the public meeting on September 22, 1998 [item 15 (a), Ref. 8], Westinghouse stated that the default input values for the probability of detection (POD) curves are consistent with the default input values for pc-PRAISE. The revised WCAP will emphasize that the SRRA code user must ensure that the specified input values for POD are appropriate for the type of material, inspection method, component geometry, and degradation mechanism being evaluated. The staff finds this response acceptable since (POD curves are consistent with the default input values for pc-PRAISE code which has been validated and accepted by the staff for various applications. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above. In addition, the detection probabilities used in SRRA calculations should be justified and documented as part of plant specific submittals.

A.20 Service Environment

Service environments (characterized by pressure, temperature, water chemistry, flow velocity, etc.) can affect corrosion rates and crack growth rates. These effects must be addressed on a segment-by-segment basis in probabilistic structural mechanics model since the classification of high-safety-significance and low-safety-significance is based on a segment-by-segment basis.

The SRRA code allows the effects of service environment to be included in calculations of piping failure probabilities. For the failure mechanism of fatigue crack growth, the equations for predicting crack growth rates are appropriate since they have been derived on the basis of tests performed with specimens exposed to reactor water environments.

Crack growth rates (for stress corrosion cracking) and wall thinning rates (for flow-assisted corrosion) can be specified by the inputs in a manner that includes appropriate effects of operating environments. Crack growth rates are appropriate since the SRRA code has incorporated bounding rates for these two degradation mechanisms, bounding rates are founded on laboratory data and service experience corresponding to high failure probabilities, and the user should specify numerical factors to be applied to these bounding rates, with the assigned factors derived from plant operating experience and engineering judgement.

In summary, the SRRA code provides an acceptable method to account for the effects of the operating environment since the method is largely reliant on qualitative judgments to indirectly assign quantitative factors. This is appropriate since typical calculations must often be performed without detailed knowledge of such factors as water chemistries and flow velocities and the documentation for the code acknowledges limitations of the approximate methodology

and recommends other methods for use as needed.

A.21 Fatigue Crack Growth Rates

The equations used by the Westinghouse SRRA code to predict fatigue crack growth rates in both stainless and ferritic steels are the same equations used by the pc-PRAISE code. These equations represent the best available correlations for the statistical distributions of mean crack growth rates and for crack growth. On the basis of this review, the staff concludes that the SRRA code has an acceptable basis for simulating fatigue crack growth rates.

A.22 IGSCC Crack Growth Rates

The equations used in SRRA to relate crack tip stress intensity factors to growth rates for stress corrosion cracks are consistent with NRC staff evaluations of BWR piping performed in the 1980s. These equations provide an acceptable approach to predict bounding growth rates for sensitized stainless steel welds in BWR water environments.

The equations implemented in the SRRA code do not provide a mechanistic basis to address stress corrosion cracking under less aggressive conditions. Limitations of the equations are acknowledged in the code documentation provided in WCAP-14572, Revision 1, Supplement 1. A code user is guided to apply knowledge of the materials/welding variables and of the plant operating conditions in combination with engineering judgement to estimate crack growth rates relative to the bounding rates incorporated into the SRRA code. The user is also guided in this difficult task with the option to assign a high, medium, or low category for the crack growth rates. With this option the code internally assigns the numerical parameter which is applied as a multiplying factor to the bounding crack growth rates.

A.23 Wall Thinning Rates

The Westinghouse SRRA code estimates wall thinning rates using a statistical correlation (mean of 0.0095 inch per year and standard deviation of 0.893 inch per year) of field measurements of thinning rates from piping subject to flow-assisted corrosion. These measured rates were from selected piping locations which had sufficient wall thinning to violate minimum wall thickness requirements and thus result in replacement of the piping.

The user of the code must apply knowledge of the piping materials, operating conditions, and (if possible) plant-specific measurements of thinning rates to assign each pipe location to the categories of high, medium, and low thinning rates. The high category corresponds to the statistical data correlation contained in the code, with the other categories corresponding to internally assigned multiples of this reference thinning rate.

Plant technical staff will typically have data available from existing programs for augmented inspection and the management of wall thinning for piping systems at their plants. In these cases, the user can override the parameters corresponding to the three standard categories,

and directly assign input to describe the best estimate and uncertainty in the thinning rates. These assignments can be based on location specific wall thickness measurements, predictions of thinning rates such as by the CHECKWORKS code, or can be based on other sources of knowledge and/or engineering judgement.

With proper inputs, the code provides a useful tool to assist in estimating piping failure probabilities attributable to wall thinning. Before issuing this SER, the staff had identified an open item that Westinghouse should expand the code documentation to provide additional guidance for selecting the input for the calculation. In the public meeting on September 22, 1998 [item 15(b), Ref. 8], Westinghouse stated that the next Revision of WCAP-14572, Supplement 1, will provide detailed guidelines for simplified input variables and any associated assumptions that could be important in assigning the input values for the SRRA code. WCAP-14572 will also state that if more than one degradation mechanism is present in a given segment, the limiting input values for each mechanism should be combined so that a limiting failure probability is calculated for risk ranking. The staff finds the guidance in item 15(b), Ref. 8 to be acceptable because it provides sufficient guidance for the code user for selecting input parameters. The staff's approval is conditioned upon Westinghouse making the change to WCAP-14572 described above.

A.24 Material Property Variability

Variability and uncertainties in certain material properties have a large influence on calculated failure probabilities. Nonetheless it is appropriate for probabilistic structural mechanics codes to treat some material properties as deterministic, while the variability and uncertainty in other properties must be simulated in the probabilistic model. Experience has shown that it is critical to treat the material input parameters associated with crack growth rates, fracture toughness, and strength levels as random variables.

The SRRA code treats probabilistically the important parameters which describe material properties. The staff finds that the code provides an acceptable basis to account for uncertainties in material-related characteristics since the code documentation clearly indicates which material properties are treated in a probabilistic manner and which parameters are treated as deterministic inputs.

A.25 SUMMARY AND CONCLUSIONS

This review concludes that the Westinghouse SRRA code provides an acceptable method that can be used, in combination with trends from data bases and insights from plant operating experience, for estimating piping failure probabilities. The underlying deterministic models used by the code are based on sound engineering principles and make use of inputs which are within the knowledge base of experts that will apply the code. Effects of variability and uncertainties in code inputs are simulated in a reasonable manner. The documentation for the SRRA computer code shows examples where the code has been benchmarked against other computer codes and validated with service experience.

While the SRRA code can be applied as a useful tool for estimating piping failure probabilities,

the present review has identified a number of limitations in the types of calculations that can be performed by the code. Some of the concerns which users of the code must be aware include:

- The quality and usefulness of results from the SRRA code are very dependent on the quality of inputs provided to the code. It is important that users of SRRA be adequately trained in the features and limitations of the code, and have the access to detailed information of the plant specific piping systems being modeled.

- The results of SRRA calculations should always be reviewed to ensure that they are reasonable and consistent with plant operating experience. Data from plant operation should be used to review and refine inputs to calculations. In all cases, greater confidence should be placed in relative values of calculate failure probabilities than on absolute values of these probabilities.

- The stresses used for plant specific applications should be based on actual plant experience and operational practices (including thermal and vibrational fatigue stresses), which may differ from the stresses used for purposes of the original design of the plant.

- The present review describes some numerical difficulties and issues encountered in comparing break probabilities for the fatigue of stainless steel piping when leak detection was included in the calculations. Nevertheless, the present review agrees with the overall conclusion as stated by Westinghouse, that the calculations did successfully benchmark the calculations for the small leak, large leak, and full break probabilities.

- The simplified nature of the SRRA code has resulted in a number of conservative assumptions and inputs being used in applications of the code. It is therefore recommended that sensitivity calculations be performed to ensure that excessive conservatism does not unrealistically impact the categorization and selection of piping locations to be inspected.

- The model of piping fatigue and stress corrosion cracking by the SRRA code addresses only failures due to the growth of preexisting fabrication flaws and does not address service induced initiation of cracks. Given plant operating experience which shows that piping failures by fatigue and IGSCC are very often due to initiated cracks, the prediction of failure probabilities for these degradation mechanisms will often be better addressed by other methods and/or other computer codes, such as pc-PRAISE

- The SRRA model for flow assisted corrosion and wastage only addresses the variability in wall thinning rates, and assumes that the user has a basis for assigning values for expected or nominal thinning rates. Application of the SRRA model should be made within the context of existing plant programs for the inspection and management of wall thinning of piping systems. The SRRA code can be applied most effectively if there are means to estimate the thinning rates, based, for example, on data collected from wall thinning measurements or from predictions of computer codes such as the EPRI developed code CHECKWORKS.

- The pilot applications of the SRRA code to risk-informed ISI as described in WCAP-14572 represent a new and evolving application of the probabilistic structural mechanics technology. Lessons learned from the pilot applications and consideration of the code limitations as identified in the present review should be used to guide the future development and

enhancement the SRRA code.

A.26 REFERENCES for APPENDIX A

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