9C

NUREG/CR-3869
PNL-5149

Analyses of the Impact of Inservice Inspection Using a Piping Reliability Model

Prepared by F. A. Simonen, H. H. Woo

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory Commission

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations,* and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $4.25

NUREG/CR-3869
PNL-5149
R5

Analyses of the Impact of Inservice Inspection Using a Piping Reliability Model

Manuscript Completed: July 1984
Date Published: July 1984

Prepared by
F. A. Simonen, H. H. Woo*

Pacific Northwest Laboratory
Richland, WA 99352

*Lawrence Livermore National Laboratory

Prepared for
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
NRC FIN B2289

ABSTRACT

This report presents the results of a study of the impact of inservice inspection (ISI) programs on the reliability of specific nuclear piping systems that have actually failed in service. Two major factors are considered in the ISI programs: one is the capability of detecting flaws; the other is the frequency of performing ISI. A probabilistic fracture mechanics model is used to estimate the reliability of two nuclear piping lines over the plant life as functions of the ISI programs. Examples chosen for the study are the PWR feedwater steam generator nozzle cracking incident and the BWR recirculation line safe-end cracking incident. The results show that an effective inservice inspection requires a suitable combination of flaw detection capability and inspection schedule. An augmented inspection schedule is required for piping with fast-growing flaws to ensure that the inspection is done before the flaws reach critical sizes. Also, the elimination of "poor" inspection teams through training and qualification testing can produce significant benefits to ISI effectiveness.

CONTENTS

CONTENTS cont

FIGURES

FIGURES cont

TABLES

ANALYSES OF THE IMPACT OF INSERVICE INSPECTION USING A PIPING RELIABILITY MODEL

1.0 INTRODUCTION

This report describes probabilistic fracture mechanics calculations that predict the impact of inservice inspection on the growth of cracks in nuclear piping. These calculations are one of a series of evaluations being performed at Pacific Northwest Laboratory (PNL) for the U.S. Nuclear Regulatory Commission (NRC) as part of the research program "Integration of NDE Reliability and Fracture Mechanics" (NUREG/CR-3059; NUREG/CR-3176; Heasler and Simonen 1983). Data generated on this program are intended to provide a basis on which to formulate revisions to both the ASME Section XI Boiler and Pressure Vessel Code and regulatory requirements needed to assure low failure probabilities of reactor components. The primary objective of PNL's research program has been to determine the reliability of ultrasonic inservice inspection (ISI) as performed on commercial, light water reactor primary systems. This report concerns a second objective, which is to determine the impact of nondestructive evaluation (NDE) unreliability on system safety and to determine the level of inspection reliability required to assure a suitably low failure probability.

The calculations described in this report were performed as a cooperative effort between PNL and Lawrence Livermore National Laboratory (LLNL). Under NRC sponsorship LLNL has been participating in piping reliability studies since 1980. The development of the PRAISE code (Piping Reliablity Analysis Including Seismic Events) was part of the LLNL effort (NUREG/CR-2189). This code formed the basis for various case studies in which the probabilities for both leak and break of various piping systems were calculated (NUREG/CR-2301; NUREG/CR-2801; Woo and Chou 1982). The objective of the PNL/LLNL cooperative effort was to extend the existing piping reliability models and then to apply these models using, as inputs, the crack detection data from the Piping Inspection Round Robin at PNL.

Previous work at PNL (Heasler and Simonen 1983) evaluated the impact of ISI on the reliability of primary coolant loop piping of a pressurized water reactor (PWR). The LLNL model of the Zion Reactor was applied, for which the calculated failure probabilities were relatively low. In contrast, the current calculations were performed to evaluate the impact of ISI on systems with high failure probabilities.

The selected models described two actual failure incidents [PWR feedwater line and boiling water reactor (BWR) recirculation line]. In each case, the relative importance of flaw detection capability and inspection schedule was evaluated by simulating alternative inspection scenarios.

This report begins by describing the detection capabilities from PNL's round robin and how this information was applied in the current calculations. Sections 3 and 4 provide an overview of the piping reliability model and details of the calculations for the PWR feedwater line and BWR recirculation line cracking incidents. Section 5 presents an interpretation of the case study results. The overall conclusions are provided in Section 6. An appendix relates the results obtained in this study to results of other calculations for which the analytic models and characteristics of the systems themselves differed from those described in this report.

2.0 ROUND ROBIN FLAW DETECTION CAPABILITY DATA

The piping inspection round robin conducted at PNL for the NRC was aimed at determining the effectiveness of inservice ultrasonic inspection of nuclear system piping (NUREG/CR-1696, Vol.1). The round robin included three materials and six test teams. The measured inspection effectiveness encompassed 1) minimum ASME code requirements, 2) as-practiced field procedures, and 3) improved procedures. Inspections were made under laboratory and simulated field conditions with flaws located on both the near and far side of the weld. The following subsections present the probability of crack detection (POD) curves used in the analyses.

2.1 PROBABILITY OF CRACK DETECTION

The probability of crack (or flaw) detection (POD) is defined as the probability that a crack with certain specified physical characteristics (e.g., size) is found during an inspection and correctly classified as a crack. Probability of crack detection is measured directly by summing the number of times a crack of a given category is successfully detected and then dividing by the total number of times the crack is subject to inspection.

The normal cumulative distribution function, Φ (Mood, Graybill and Boes 1974, pp. 107-108), was used to fit the round robin data. The functional form is

$$POD = \Phi(x) = \int_{-\infty}^{x} \frac{1}{\sqrt{2\pi}} e^{-t^2/2} dt, \text{ and } x = U + B \ln \frac{a}{t} \qquad (2.1)$$

where U and B are constants related to pipe materials, ultrasonic inspection procedures, and inspection access. The dimensions a and t are crack depth and pipe wall thickness, respectively. The probability of crack nondetection, P_{ND}, is defined as

$$P_{ND} = 1 - POD \qquad (2.2)$$

Three POD curves identified as "poor", "good", and "advanced" teams were defined as follows (NUREG/CR-2189):

- poor team - This curve represents a lower bound on performance among round robin teams. This team did follow minimum ASME code requirements.
- good team - This curve represents the better teams in the round robin.
- advanced team - This curve represents the performance that may be achieved with improved procedures and existing technology. It assumes a 0.9999 flaw detection probability for a through-wall flaw, and a

probability of detection of about 90 percent for a flaw with a depth of 10 percent of the wall.

The next two subsections present the POD curves in the form of Equation (2.1) for both ferritic steel and stainless steel materials.

2.1.1 Ferritic Steel POD Curves

Three POD curves (NUREG/CR-2716, Vol. 3) were generated by fitting round robin data for ferritic steel under the conditions where personnel followed the ASME code procedures and had near-side access for their inspection.

- poor team - $\text{POD} = \Phi\left[0.432 + 0.163 \ln(a/t)\right]$ (2.3)
- good team - $\text{POD} = \Phi\left[1.75 + 0.583 \ln(a/t)\right]$ (2.4)
- advanced team - $\text{POD} = \Phi\left[3.63 + 1.106 \ln(a/t)\right]$ (2.5)

These functions apply to flaw depths greater than 5 percent of wall thickness. A detection capability of POD = 0.0 for a/t = 0.0 was assumed, and a linear interpolation with a/t was used between 0 and 5 percent of wall thickness. Some numerical values for ferritic steel POD are presented in Table 2.1.

TABLE 2.1. Numerical Values for Ferritic Steel Probability of Crack Detection Curves

	POD		
a/t	Poor	Good	Advanced
0.0	0.0	0.0	0.0
0.005	0.048	0.045	0.062
0.01	0.096	0.1	0.125
0.02	0.191	0.2	0.25
0.03	0.287	0.3	0.375
0.05	0.487	0.5	0.624
0.10	0.522	0.655	0.86
0.25	0.572	0.826	0.982
1.00	0.666	0.96	0.9999

2.1.2 Stainless Steel POD Curves

Three POD curves for 10-inch stainless steel piping with stress corrosion cracks (Doctor et al. 1983) were fitted to the round robin data. The conditions for performing ultrasonic inspections were that inspectors had near-side access and followed field inspection procedures.

- poor team - POD = $\Phi[0.24 + 1.485 \ln(a/t)]$ (2.6)
- good team - POD = $\Phi[1.526 + 0.533 \ln(a/t)]$ (2.7)
- advanced team - POD = $\Phi[3.63 + 1.106 \ln(a/t)]$ (2.8)

Again, these functions apply for flaw depths greater than 5 percent of the wall thickness. A detection capability of POD = 0.0 for a/t = 0.0 was assumed, and a linear interpolation with a/t was used between 0 and 5 percent of wall thickness. Table 2.2 gives some numerical values for stainless steel POD curves.

TABLE 2.2. Numerical Values for Stainless Steel Probability of Crack Detection Curves

	POD		
a/t	Poor	Good	Advanced
0.0	0.0	0.0	0.0
0.005	0.0	0.047	0.062
0.010	0.0	0.094	0.125
0.020	0.0	0.189	0.25
0.030	0.0	0.287	0.375
0.050	0.0	0.472	0.624
0.1	0.001	0.617	0.86
0.2	0.016	0.748	0.968
0.4	0.0131	0.85	0.9956
0.7	0.386	0.909	0.9994
1.0	0.595	0.936	0.9999

3.0 PIPING RELIABILITY MODEL OVERVIEW

The piping reliability model was developed on the basis of probabilistic fracture mechanics concepts. The computational procedure for the estimation of leak probability combines various random variables, such as initial crack size distribution, flaw (or crack) detection probability, crack growth relation, and the deterministic stress history. As indicated by Figure 3.1, the computation starts with the initial size of crack-like defects (i.e., flaws) at a given location. These growing cracks are detected with a certain probability during preservice and inservice inspections. Cracks that escape detection and repair can grow following subcritical crack growth characteristics such as fatigue crack growth and stress corrosion cracking. The critical crack size for leak can be defined by using an appropriate criterion (e.g., through-wall cracking). The probability of leak at the pipe location analyzed is equal to the probability of a crack growing to corresponding critical size within the specified time. The Monte Carlo method was used in the numerical simulation. It is obvious that crack detection capability and inspection time influence the leak probability results because they are the last elements to prevent pipe leak once the crack grows in the simulation.

Seven variables are required in the model's calculations:

- pipe material and properties
- pipe geometry, pipe cross section dimensions
- initial crack depth distribution
- loadings and associated occurrence rates - Loadings may include pipe internal pressure, dead weight, thermal restraint load, residual stress, vibratory stress, and seismic load. Occurrence rates for different loadings can be specified.
- crack growth models - $da/dn = f_1(C, m, \Delta K)$ for fatigue crack growth and $da/dt = f_2(C, m, K)$ for stress corrosion cracking where C and m are material constants; K and ΔK are the applied stress intensity factor and its range, respectively; n is the number of cycles, and t is the time variable.
- detection probability models for cracks and leaks
- inservice inspection schedules.

The next section describes how the piping reliability model was used to demonstrate the impact of flaw detection capability on the reliability of nuclear piping.

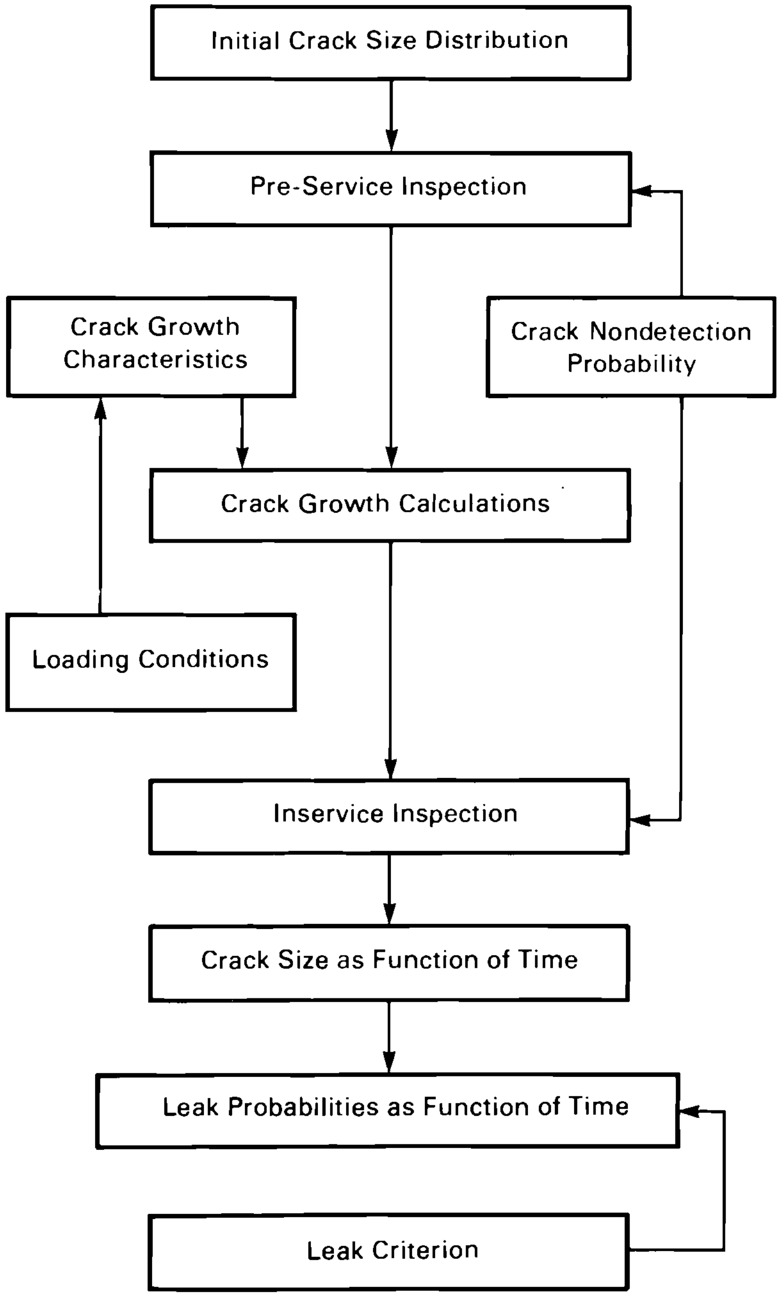

FIGURE 3.1. Computational Procedure Used in Estimating Leak Probability for Nuclear Piping

4.0 PIPING RELIABILITY MODEL APPLICATIONS

Two actual piping leak incidents were chosen to demonstrate the impact of flaw detection capability on the reliability of nuclear piping. The piping reliability model described in Section 3 was applied to a PWR feedwater line cracking incident and to a BWR recirculating line cracking incident. Two plants, identified as A and B, were studied for the first incident. One plant was studied for the second incident.

In all three model applications, the variable for initial crack size distribution at the beginning of the simulation was assigned a fixed or deterministic value. The initial crack depth was determined such that the specified crack size at the specified time would be identical to the one observed during the incident. Mean-value curves for crack growth rates were used in determining the initial crack size. However, in the real simulation, a probabilistic crack growth model was used. Three flaw detection probability curves in conjunction with various inspection schedules were considered. A leak due to through-wall cracking was defined as the failure criterion. The probabilities of leak were then expressed as a function of plant life.

This section describes the computations used and the results obtained when the model was applied to the two cracking incidents.

4.1 PRESSURIZED WATER REACTOR FEEDWATER LINE CRACKING INCIDENT

4.1.1 Background

On May 20, 1979, the Indiana and Michigan Power Company informed the NRC of cracking in two feedwater lines at D.C. Cook Unit 2. Circumferential through-wall cracks were detected at the 16-inch lines in the junction of the feedwater lines and steam generator. Subsequent volumetric examination by radiography revealed crack indications at similar locations in all feedwater lines of both Units 1 and 2. As a result of this incident, the NRC Office of Inspection and Enforcement issued I.E. Bulletin 79-13 requiring inspection of all PWR feedwater lines. Inspections through March 1980 revealed pipe cracks or fabrication defects requiring repair in the vicinity of the feedwater nozzles at 16 of 35 PWR plants (NUREG-0691).

Extensive studies, including metallurgical analysis, in-plant instrumentation, and thermohydraulic modeling, led to the conclusion that the primary cause of cracking was a fatigue mechanism induced by thermal stratification during low-flow auxiliary feedwater injections (Westinghouse Electric Corporation 1980). Thermal stratification usually occurs during hot-standby, startup, and shutdown conditions. During such conditions, horizontal portions of the feedwater line are subjected to large temperature differences between the top and bottom of the pipes. This phenomenon is illustrated in Figure 4.1. These stratified temperature conditions vary rapidly during low-power operations and can induce high cyclic thermal stresses in the feedwater nozzle where cracking has occurred.

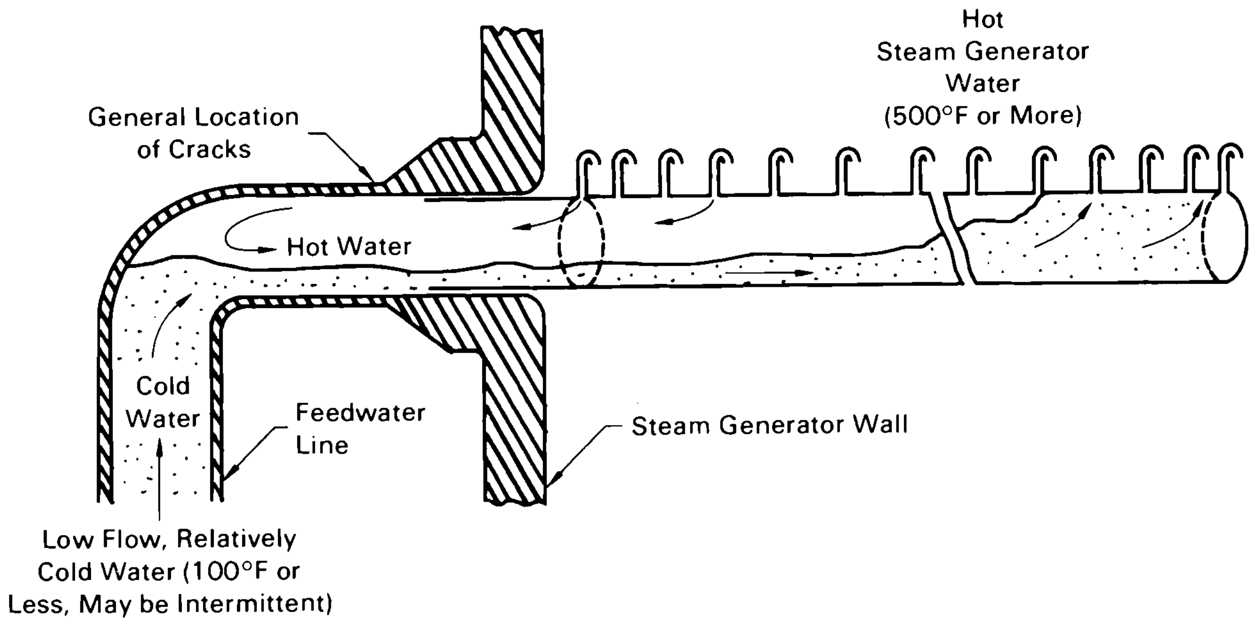

FIGURE 4.1. Thermal Stratification Phenomenon in PWR Feedwater Lines

4.1.2 Data Input to Model

The values of the variables input to the piping reliability model for simulating the feedwater line cracks are documented in the next seven subsections.

4.1.2.1 Pipe Material

The feedwater piping consisted of A106 GrB. The nozzle examined was SA508 Class 2.

4.1.2.2 Pipe Geometry

The feedwater piping was 16-inch Schedule 80. The nozzle was 16-inch Schedule 60.

4.1.2.3 Initial Crack Depth

The calculated initial crack depths for both plants A and B are listed in Table 4.1.

4.1.2.4 Crack Growth Model

The reference fatigue crack growth law in Section XI of the ASME Code for carbon and low-alloy ferritic steels was modified to represent the fatigue

TABLE 4.1. Initial Crack Depth for PWR Feedwater Line Reliability Analysis

Plant	Time of Inspection[(a)]	Wall Thickness (inch)	Observed Crack Depth (inch)	Calculated Initial Crack Depth (inch)
A	11 months	0.57	0.57	0.021
B	33 months	0.875	0.238	0.057

(a) Time from first commercial service to inspection due to feedwater line cracking incident.

crack behavior of the feedwater line piping material. The modification was done by adding a factor, P, to the reference crack growth law. The model has the form

$$\frac{da}{dn} = P\ C(\Delta K)^m \tag{4.1}$$

where P is lognormally distributed (μ the mean value, and σ the standard deviation). The randomness of P reflects the distribution of test data for the material crack growth characteristics (Bamford 1980).

The value of ΔK is the difference between the maximum and minimum values of crack tip stress intensity factors (K_I) resulting from the changing stresses during a given stress cycle. For the current analysis, a linearized stress field is assumed for stresses through the pipe wall thickness. The expression below is used to calculate K_I (Bamford, Thurman and Mahlab 1980).

$$K_I = \pi a\ (A_0 F_1 + \frac{2a}{\pi}\ A_1 F_2) \tag{4.2}$$

where a = crack depth

A_0 = axial stress at inside surface

$A_1 = \dfrac{\sigma_0 - A_0}{t}$

σ_0 = axial stress at outside surface

t = wall thickness

$F_1 = 1.126 + 0.234\ a/t + 2.20\ (a/t)^2 - 0.208\ (a/t)^3$

$F_2 = 1.073 + 0.267\ a/t + 0.666\ (a/t)^2 + 0.635\ (a/t)^3$.

The stress profile is represented by the linearization

$$\sigma = A_0 + A_1 x \qquad (4.3)$$

where x is the radial distance measured from the inside surface of the pipe.

Denoting R as the ratio of K_{min} and K_{max}, Table 4.2 shows various values of C, m, μ, and σ for different ranges of R and ΔK.

4.1.2.5 Loading Conditions

The transients used as a design basis for the feedwater lines are listed in Table 4.3, along with the number of their expected occurrences over the 40-year design lifetime of a typical PWR plant. The transients are assumed to occur at a uniform rate over the entire lifetime. For each transient, the pressure and temperature changes are used to calculate stresses. The results are presented in Table 4.3 (NUREG-0691). Results of high cycle thermal stresses resulting from thermal stratification phenomena for feedwater lines at plants A and B are presented in Tables 4.4 and 4.5, respectively (NUREG/CR-2801). In Tables 4.4 and 4.5, σ_m and σ_b are defined as membrane and bending stresses, respectively. The relationship between σ_m, σ_b, A_0 and A_1 [see Equation (4.2)] can be expressed as:

$$A_0 = \sigma_m + \sigma_b$$
$$A_1 = -2\,\sigma_b/h \qquad (4.4)$$

TABLE 4.2. Constants Used in Fatigue Crack Growth Model for Carbon and Low Alloy Steels

Ranges for R and ΔK	C	m	μ	σ
	Constants			
$R \le 0.25$				
$\Delta K^{(a)} < 19$	1.02×10^{-6}	5.95	-0.408	0.542
$\Delta K > 19$	1.01×10^{-1}	1.95	-0.408	0.542
$R \ge 0.65$				
$\Delta K < 12$	1.2×10^{-5}	5.95	-0.367	0.817
$\Delta K \le 19$	2.52×10^{-1}	1.95	-0.367	0.817
$0.65 > R > 0.25$				
$\Delta K < 12 + 7W^{(b)}$	$1.2 \times 10^{-5}W + 1.02 \times 10^{-6}W'$	5.95	$-0.367W - 0.0408W'$	$0.817 + 0.542W'$
$\Delta K \ge 12 + 7W$	$2.52 \times 10^{1}W + 1.01 \times 10^{-1}W'$	1.95	$-0.367W - 0.0408W'$	$0.817 + 0.542W'$

(a) ΔK in ksi$\sqrt{in.}$

(b) $W = (R - 0.25)/0.4$, $W' = 1 - W$.

TABLE 4.3. Stress Results Used in Fatigue Crack Growth Calculation for Design Transients

Design Transients	No. of Cycles in 40 years	Axial Stress in ksi Inside Surface		Axial Stress in ksi Outside Surface	
		Maximum	Minimum	Maximum	Minimum
Hot standby	18,300	4.77	3.67	4.77	3.67
Unit load-unload 5% per minute	18,300	5.07	3.04	4.01	4.66
Small stepload decrease	2,000	4.53	3.33	3.83	3.83
Large stepload decrease	200	8.04	3.83	1.40	3.83
Loss of power	40	17.27	3.66	-4.9	3.66
Partial loss of flow	30	17.41	3.37	-1.34	3.66
Loss of load	80	17.7	3.76	-1.04	4.76
Reactor trip	400	26.56	2.25	-7.34	4.37
Secondary side hydrotest	5	5.68	4.37	5.68	4.37

4.1.2.6 Crack Detection Probability

Three crack detection probability curves as described in Equations (2.3) through (2.5) were used for the feedwater line piping material. Those curves are referred to as the poor, good, and advanced teams.

4.1.2.7 Inservice Inspection Schedules

Table 4.6 presents six scenarios for inspection schedules. Inspections were assumed to be performed at the end of the years indicated in Table 4.6.

4.1.3 Results and Discussion

Cumulative leak probabilities for both plants A and B over a 40-year plant life are expressed as functions of inspection schedule and crack detection capability. Figure 4.2 shows the leak probabilities for plant A with inspection schedules 1, 2, and 3. It can be seen that the leak probability increases rapidly from zero to unity within the first 5 years when no inspection is performed. As shown in Figure 4.2, schedules 1 and 2 do not significantly help in reducing the leak probability (dropping from unity to a value of 0.9). However, schedule 3 lowers the leak probability to a value of 0.5 starting from the fifth year. Furthermore, the effect of

TABLE 4.4. Stress Results Used in Fatigue Crack Growth Calculation for Plant A for Thermal Stratification Conditions

No. of Cycles Estimated up to the Incident	Maximum Stress in ksi		Minimum Stress in ksi	
	σ_m	σ_b	σ_m	σ_b
1361	29.40	17.27	-39.85	-51.88
545	29.35	17.32	-30.53	-41.32
908	30.86	15.82	-26.45	-38.48
545	29.35	17.32	-21.84	-31.37
454	30.21	16.46	-21.40	-33.43
2086	29.35	17.32	-21.84	-31.37
727	26.92	15.75	1.65	-7.81
90	26.92	15.75	2.41	-7.81
454	20.57	11.55	2.35	-6.62
1271	20.32	8.83	2.41	-6.56
20	20.32	8.83	0.0	0.0
454	20.32	8.83	5.17	-2.28
726	20.32	8.83	7.79	3.83
1522	20.32	8.82	9.46	4.44
817	19.19	8.37	9.46	4.44
454	15.52	6.89	9.46	4.44

different inspection teams (poor, good, advanced) begins to have a significant impact on leak probability when an augmented inspection program such as schedule 3 is adopted. Similarly, Figure 4.3 shows leak probabilities for plant A with inspection schedules 4, 5, and 6. It is clear that only inspection schedule 6 has a slight impact on the leak probability; schedules 4 and 5 show no effect at all because the first inspection for both schedules is beyond the fifth year.

Figure 4.4 shows the leak probabilities for plant B over a 40-year plant life with inspection schedules 1, 2, and 3. The effect of schedules 4, 5,

TABLE 4.5. Stress Results Used in Fatigue Crack Growth Calculation for Plant B During Thermal Stratification Conditions

No. of Cycles Estimated up to the Incident	Maximum Stress in ksi		Minimum Stress in ksi	
	σ_m	σ_b	σ_m	σ_b
270	27.29	15.75	-35.90	-47.15
90	24.38	12.84	-36.52	-46.52
1170	23.75	13.46	-27.20	-37.20
90	21.93	9.46	-17.98	-26.74
540	20.07	11.31	-16.04	-24.74
180	20.77	10.61	-8.89	-17.65
360	20.31	11.08	1.79	-6.97
2340	20.07	11.31	2.98	-5.82
630	19.31	8.41	2.98	-5.72
540	17.16	7.55	2.98	-5.72
450	14.93	6.24	2.69	-5.40
40	14.64	6.53	0.0	0.0
50	14.64	6.53	5.46	-1.84
670	10.60	3.31	4.32	-0.7
2070	9.46	4.44	8.42	4.06

and 6 is shown in Figure 4.5. It is important to point out that the leak probability with no inspection does not increase as fast as it did for Plant A. Furthermore, the leak probability approaches unity around the 25th year. Because the first inspection for all six schedules is long before the 25th year, all schedules have a positive impact on the leak probabilities. In general, schedule 3 shows the greatest reduction in leak probability (to a value of less than 0.1) when compared with the no-inspection case. The results for schedules 2 and 6 are comparable; their maximum leak probabilities are between 0.2 and 0.4. Schedules 1, 4, and 5 give an improvement in reliability and reduce the leak probabilities to about 0.6, 0.9, and 0.7, respectively.

TABLE 4.6. Inspection Schedules for PWR Feedwater Line Steam Generator Nozzle

Schedule	Description	Inspection time (end of year)
1	ASME Program A[a]	3, 10, 23, 40
2	1/2 inspection intervals of ASME Program A	1.5, 3, 6.5, 10, 16.5, 23, 31.5, 40
3	1/5 inspection intervals of ASME Program A	0.6, 1.2, 1.8, 2.4, 3, 4.4, 5.8, 7.2, 8.6, 10, 12.6, 15.2, 17.8, 20.4, 23, 26.4, 29.8, 33.2, 26.6, 40
4	ASME Program B[a]	10, 20, 30, 40
5	1/2 inspection intervals of ASME Program B	5, 10, 15, ..., 40
6	1/5 inspection intervals of ASME Program B	2, 4, 6, ..., 40

(a) Refers to ASME Boiler and Pressure Vessel Code, Section XI, Inservice Inspection Programs.

Although six inspection schedules were adopted for this assessment of pipe leak at PWR feedwater line steam generator nozzles, it is obvious that some inspection schedules are not realistic (e.g., schedule 3), because inservice inspections are usually performed during scheduled outages. The six schedules selected were intended to cover a wide range of inspection schedules. Interpolation of leak probabilities resulting from these six scenarios should give a good estimate for other inspection schedules. The results shown in Figures 4.2 through 4.5 do not indicate large differences in predicted leak probabilities as a function of flaw detection capability (POD curves). This is due in part to the satisfactory performance for ferritic piping of even the "poor" team in the PNL piping inspection round robin. Another important factor is the relatively high leak probabilities for the PWR feedwater lines early in the plant life. In this situation, the inspection interval is of overriding importance. An outstanding detection capability does not offset the impact of an untimely inspection.

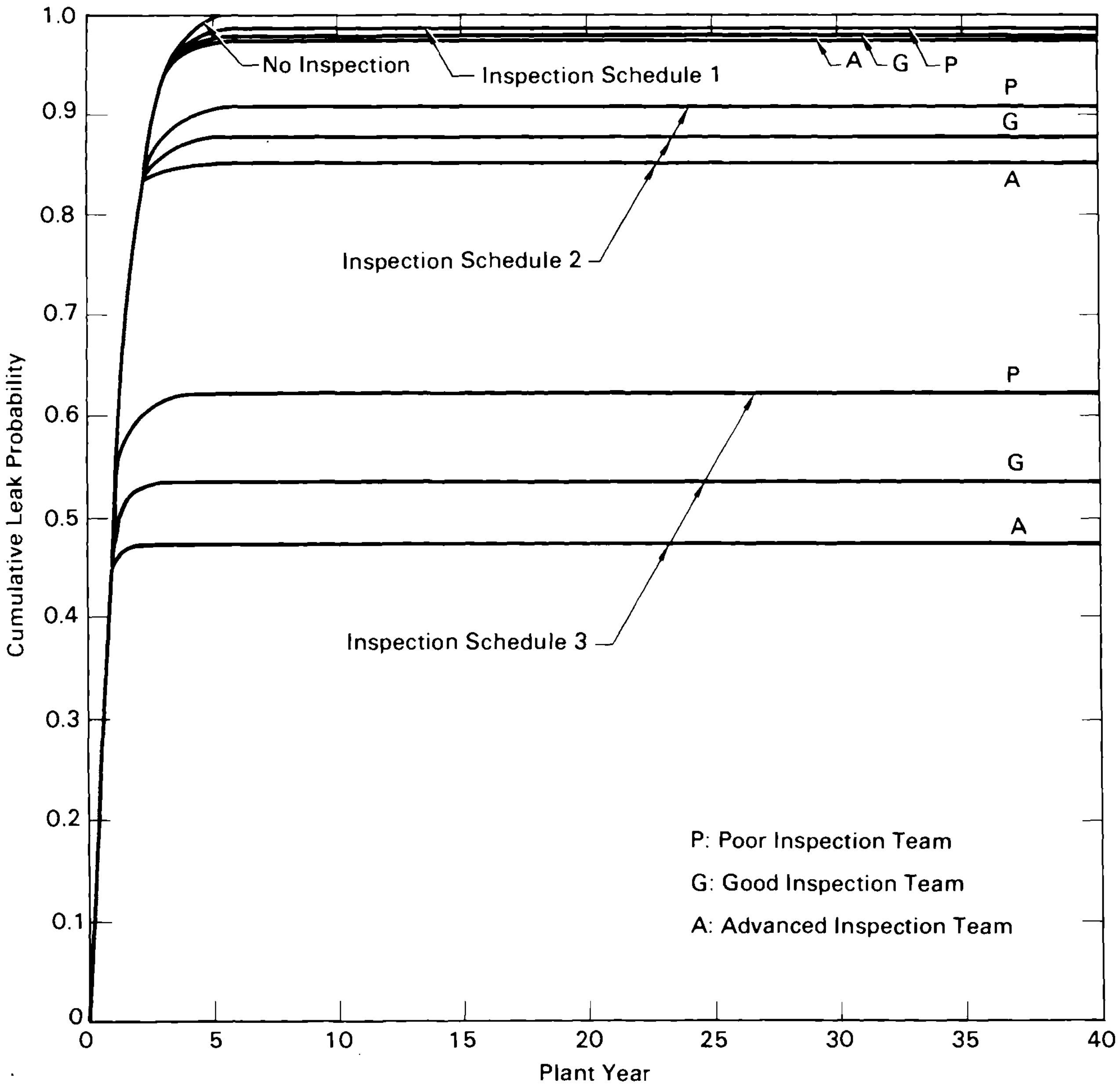

FIGURE 4.2. Cumulative Leak Probablities for PWR (Plant A) Feedwater Steam Generator Nozzle Using Three Inspection Teams and Inspection Schedules 1, 2, and 3

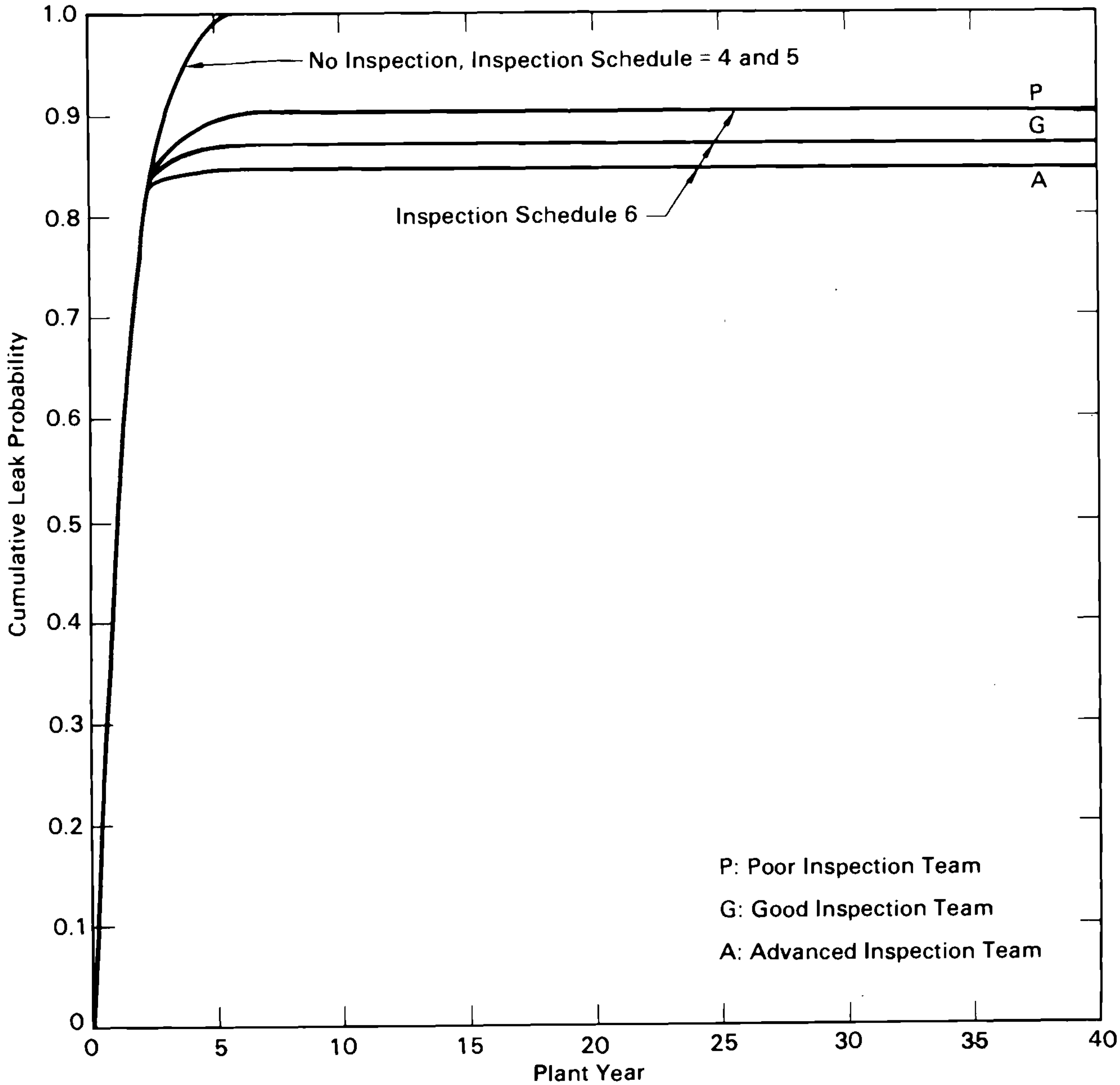

FIGURE 4.3. Cumulative Leak Probabilities for PWR (Plant A) Feedwater Steam Generator Nozzle Using Three Inspection Teams and Inspection Schedules 4, 5, and 6

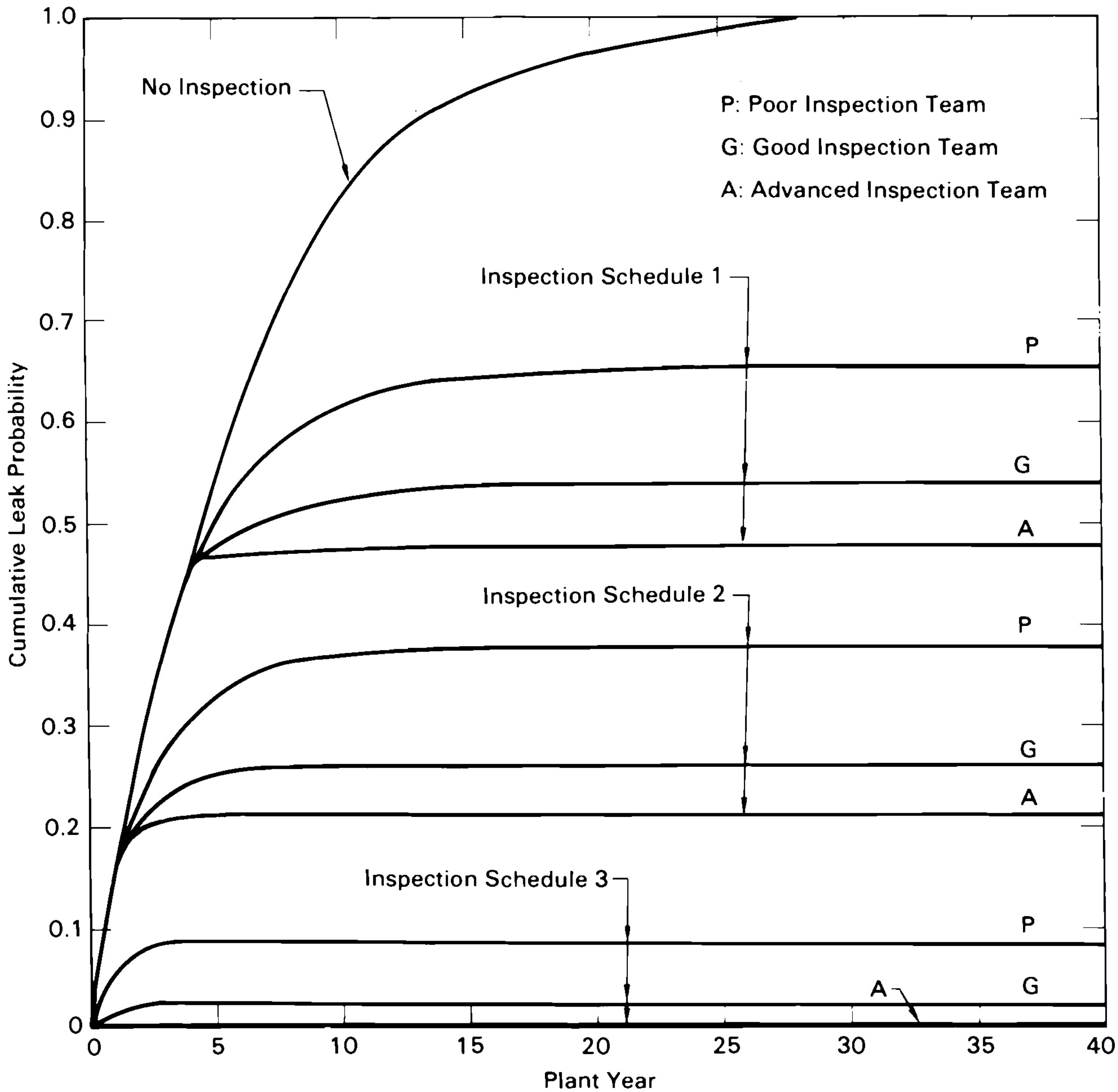

FIGURE 4.4. Cumulative Leak Probabilities for PWR (Plant B) Feedwater Steam Generator Nozzle Using Three Inspection Teams and Inspection Schedules 1, 2, and 3

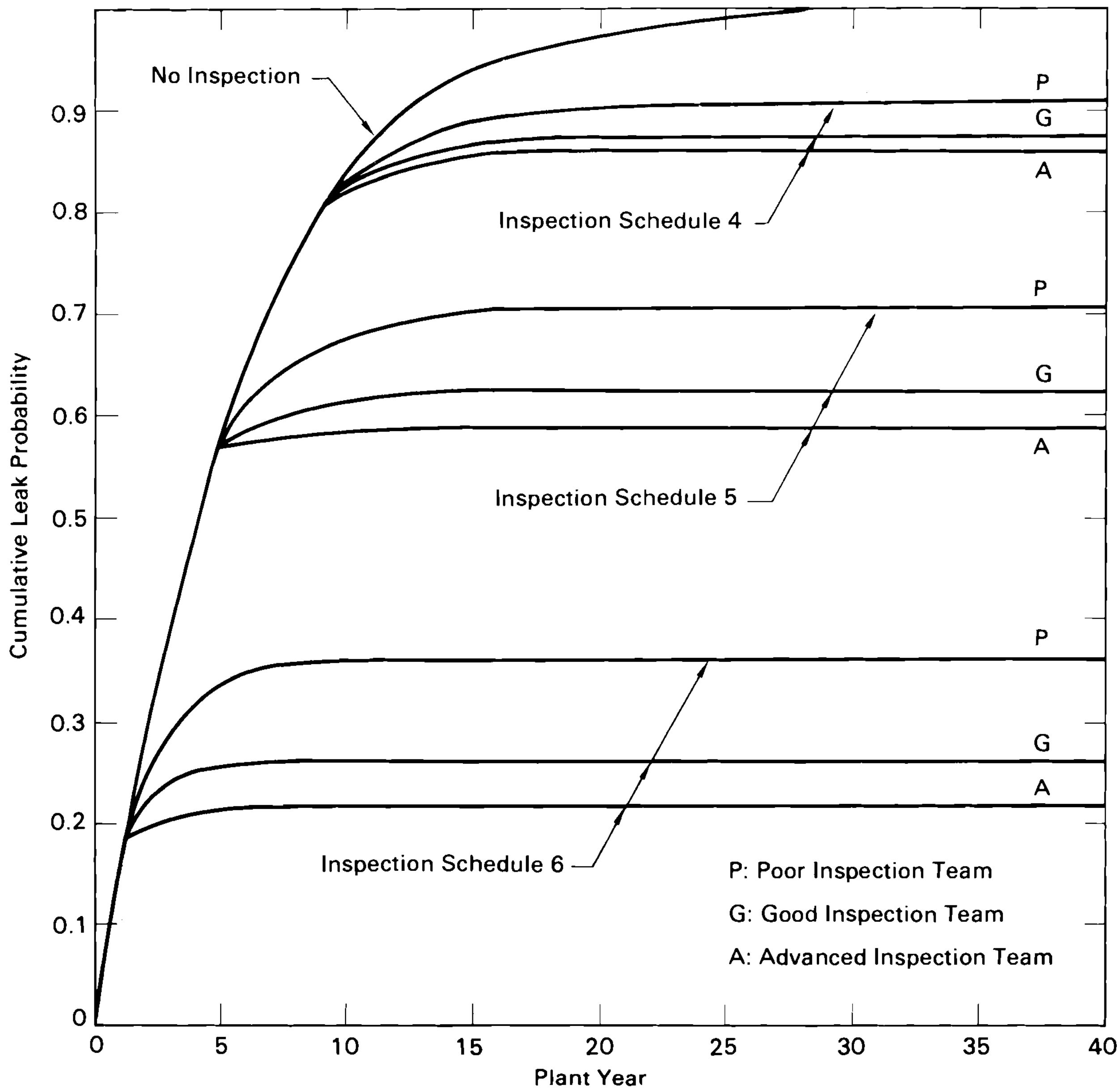

FIGURE 4.5. Cumulative Leak Probabilities for PWR (Plant B) Feedwater Steam Generator Nozzle Using Three Inspection Teams and Inspection Schedules 4, 5, and 6

4.2 BOILING WATER REACTOR RECIRCULATION LINE CRACKING INCIDENT

4.2.1 Background

In June 1978, a 3-gpm leak was discovered in one of the eight recirculation inlet nozzle safe ends at one BWR plant (NUREG-0531). The safe ends at the recirculation inlet nozzle are used to facilitate welding of the stainless steel inlet piping to the carbon steel of the reactor vessel nozzles. A thermal sleeve is welded into each safe end to direct coolant flow into the vessel. Figure 4.6 illustrates the configuration of the nozzle, safe end, piping, and thermal sleeve.

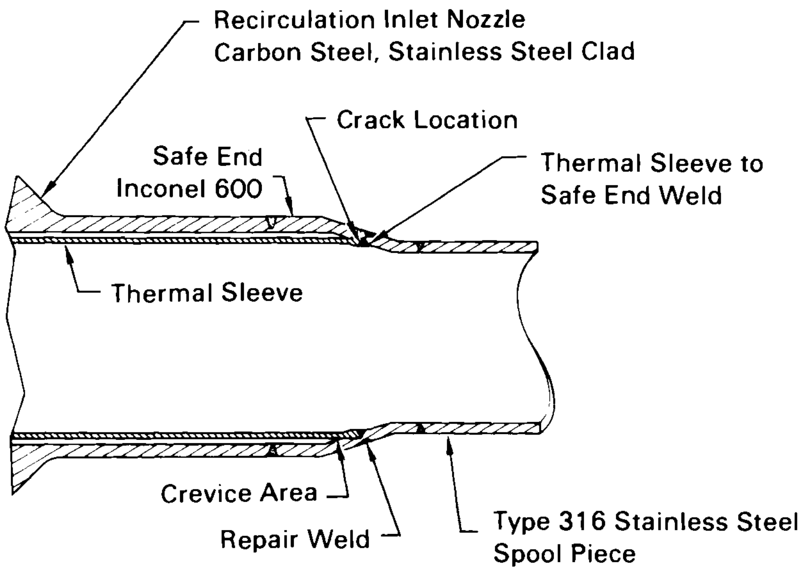

FIGURE 4.6. Recirculation Inlet Nozzle Safe End Configuration at BWR Plant

Detailed fracture analyses (Burghard 1980) led to the conclusion that crack initiation and propagation resulted from a combination of 1) the high residual stresses and operating stresses at the thermal sleeve attachment weld, 2) the oxygen in the coolant, and 3) a chemical environment, resulting from a crevice formed by the safe end thermal sleeve attachment configuration.

4.2.2 Data Input to Model

4.2.2.1 Safe End Geometry and Material

The safe end modeled had an outside diameter of 13.2 inches and was 1.15 inches thick. The material was Inconel 600.

4.2.2.2 Initial Crack Depth

The initial crack depth was specified as 0.01 inch.

4.2.2.3 Crack Growth Model

The crack growth model was based on Boyce and Woo (1983) and is expressed as

$$\frac{da}{dt} = C(K)^m \tag{4.5}$$

where a = crack depth in inches

t = time in hours

C = lognormal distribution with median value 1.44×10^{-8} and standard deviation 3.112×10

m = 1.935

$$K = \sqrt{\pi a}\left[A_0F_1 + \frac{2a}{\pi} A_1F_2 + \frac{a^2}{2} A_2F_3 + \frac{4}{3\pi} a^3 A_3F_4\right], \text{ ksi}\sqrt{\text{in}}$$

in which

$$F_1 = 1.1 + 0.9544(a/h) + 0.2920(a/h)^2$$

$$F_2 = 1.0 + 0.2979(a/h) + 0.6042(a/h)^2$$

$$F_3 = 1.0 + 0.1292(a/h) + 1.083(a/h)^2$$

$$F_4 = 1.0 + 0.009165(a/h) + 0.5584(a/h)^2$$

$$A_0 = 81.84$$

$$A_1 = -413.26$$

$$A_2 = 707.5$$

$$A_3 = -377.9.$$

It should be noted that the median value of C was determined from Equation (4.5) by assuming an initial crack depth of 0.01 inch, which was consistent with the pipe leak at the specific time. Because of the lack of test data for growth rates of stress corrosion cracks in Inconel, the standard deviation of C was assumed to be identical to the one compiled from the test data for stainless steels under a similar corrosion environment (General Electric Company 1982).

4.2.2.4 Loading Conditions

The contributing stresses are those induced by pressure, dead weight, and restraint of thermal expansion, as well as residual and peak stresses. The axial stress profile through the pipe wall can be approximately expressed by the third-order polynomial

$$\sigma(x/h) = A_0 + A_1(x/h) + A_2(x/h)^2 + A_3(x/h)^3 \tag{4.6}$$

where x/h is the normalized radial distance measured from inside the pipe wall to an arbitrary point within the wall. A_0, A_1, A_2, and A_3 are the coefficients given in Equation (4.5).

4.2.2.5 Crack Detection Probability

Because Inconel 600 piping was not part of the PNL piping inspection round robin, the crack detection probability curves for stress corrosion cracks within welds of wrought stainless steel piping were assumed to apply to Inconel 600 as well. Equations (2.6) through (2.8) give three POD curves, referred to as the poor, good, and advanced inspection reliability.

4.2.2.6 Inservice Inspection Schedules

Table 4.7 presents four schedules for inservice inspection. Inspections are assumed to be performed at the end of the year indicated in Table 4.7.

TABLE 4.7. Inspection Schedules for BWR Recirculation Line Reactor Vessel Nozzle Safe End

Scenario	Description	Inspection Schedule (end of year)
1	ASME Program B[a]	10, 20, 30, 40
2	1/2 inspection intervals of ASME Program B	5, 10, 15, ..., 35, 40
3	1/5 inspection intervals of ASME Program B	2, 4, 6, ..., 38, 40
4	1/10 inspection intervals of ASME Program B	1, 2, 3, ..., 39, 40

(a) Refers to ASME Boiler and Pressure Vessel Code, Section XI, Inservice Inspection Program

4.2.3 Results and Discussion

Cumulative leak probabilities for BWR recirculation line safe ends over a 40-year plant life are expressed as functions of the four inspection schedules and the crack detection capabilities for the three inspection teams. Figure 4.7 shows the predicted leak probability over the 40-year plant life for the poor inspection teams and the four inspection schedules.

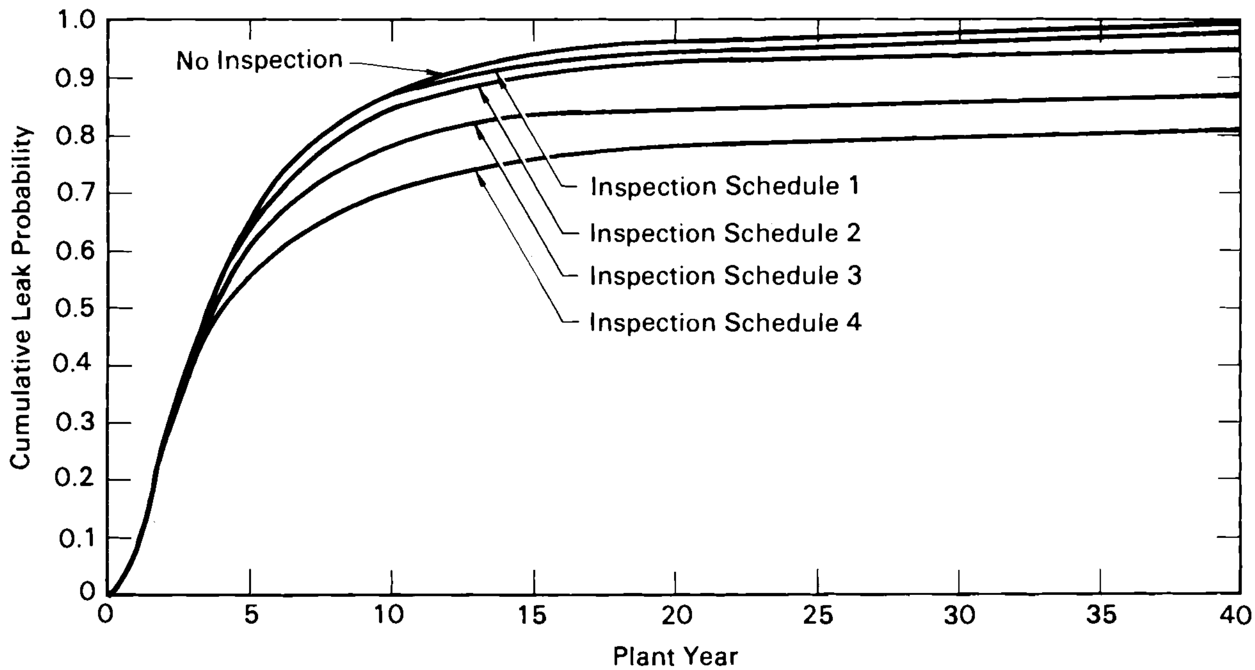

FIGURE 4.7. Cumulative Leak Probabilities for BWR Recirculation Nozzle Safe End Based on Poor Inspection Team and Four Inspection Schedules

It can be seen that, when compared with the case of no inspection, the reliability at the safe end is not improved significantly by the poor team's inspection, even with an augmented inspection program such as schedule 1. In other words, the team with the poor flaw detection capability is of no benefit in improving the reliability of this safe end, regardless of the inspection schedule chosen. However, both good and advanced inspection teams provide an improvement and reduce leak probabilities, as shown in Figures 4.8 and 4.9, respectively. A good inspection team can cut the leak probabilities for the safe end from about unity to 0.89 (schedule 4) at the end of plant life. With the help of an advanced team, the leak probabilities become 0.87 (schedule 1), 0.65 (schedule 2), 0.26 (schedule 3), and 0.06 (schedule 4) at the end of plant life.

A comparison of results shown in Figures 4.8 and 4.9 indicates that the improvement for the good inspection team is comparable to that for the advanced team. As for the PWR feedwater line example, the leak probabilities approach unity early in the plant life. Evidently, in this situation the inspection interval is of primary importance, once a certain minimal level of crack detection capability is achieved.

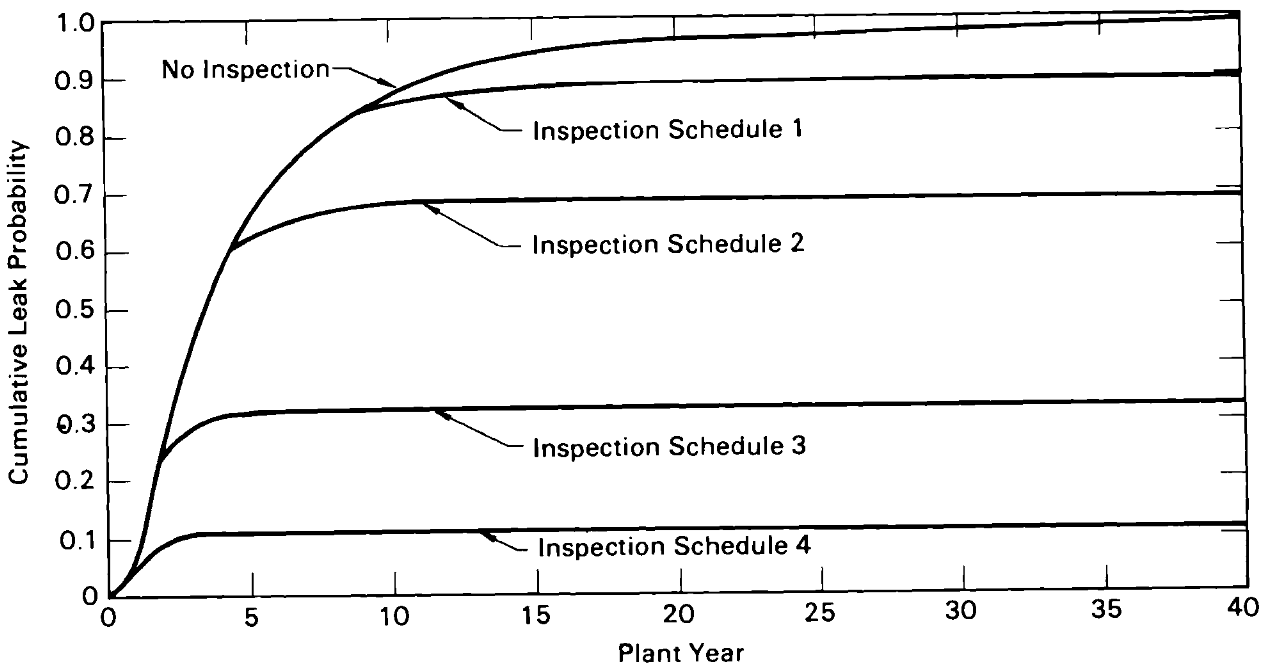

FIGURE 4.8. Cumulative Leak Probabilities for BWR Recirculation Nozzle Safe End Based on Good Inspection Team and Four Inspection Schedules

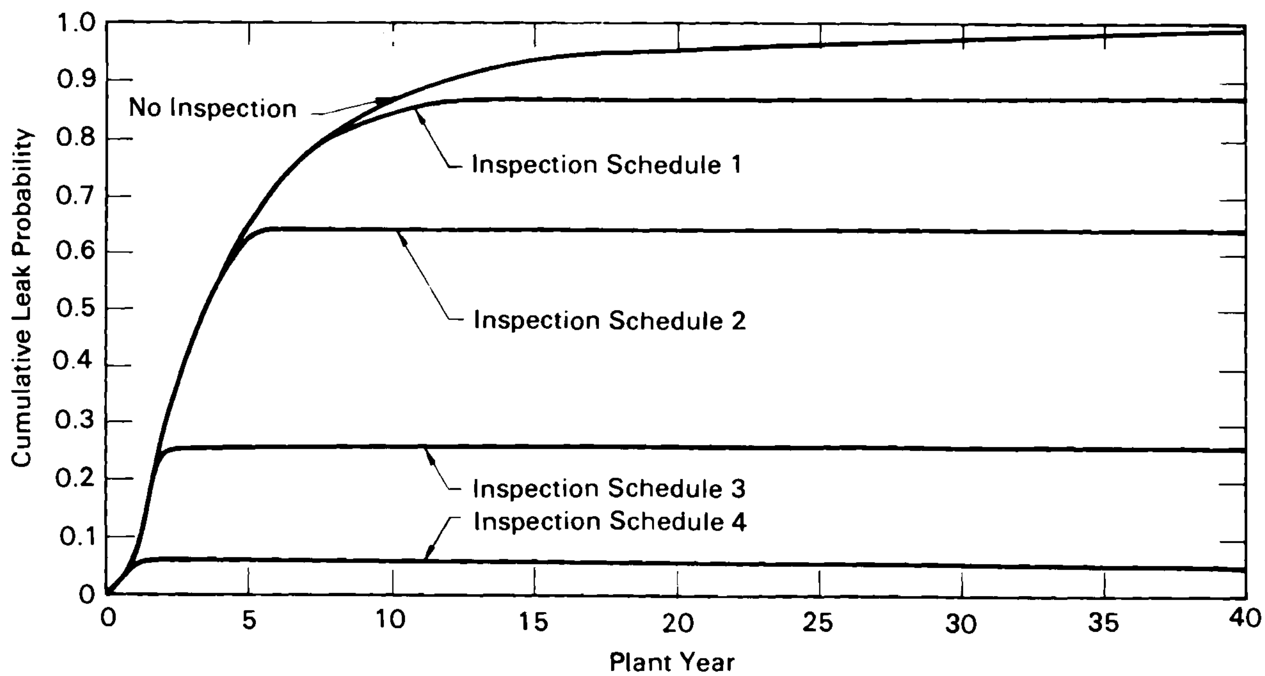

FIGURE 4.9. Cumulative Leak Probabilities for BWR Recirculation Nozzle Safe End Based on Advanced Inspection Team and Four Inspection Schedules

5.0 INTERPRETATION OF RESULTS

A long-term objective of PNL's research program is to evaluate the impact of NDE on system reliability. The PWR feedwater line and BWR recirculation line incident simulations are but one of a number of fracture mechanics calculations in which the impact of NDE is being evaluated. This section addresses concerns regarding 1) case-specific characteristics of the two cracking incidents evaluated and 2) inputs and assumptions to the analytical models. Both factors may significantly influence the trends suggested by the results of the specific probabilistic fracture mechanics analyses described in this report.

5.1 EFFECT OF SYSTEM FAILURE RATE

The case specific analyses described in this report apply to systems for which failure by leak occurs early in the life of the reactor, and then decreases with time. This trend has a significant impact on the ability of inservice inspection to improve system reliability. The conclusions of these analyses must not be applied to situations in which the failure rate may increase during the life of the reactor. The Appendix presents data on the experience with BWR piping in the U.S., where the observed rate of piping failure by leak has tended to increase during the life of the reactors.

Figure 5.1 schematically portrays the contrasting cases in which failure rates decrease and increase with time. In general, NDE is of greater benefit for situations where failures tend to occur later in life so that inspections can be performed before cracks grow to unacceptable sizes. The analyses documented in this report considered situations for which failures occur early in the life of the reactor. In contrast, calculations are described in the Appendix for which failures occur at an increasing rate later in the plant lifetime.

Figure 5.2 shows the expected impact of NDE on systems with increasing versus decreasing failure rates. Clearly, the case studies presented in this report tend to minimize the potential impact of periodic inspection because the fracture mechanics model predicts failure rates that decrease with time. In these analyses, a disproportionate number of failures occur early, very often prior to the first scheduled inspection. In contrast, for analyses in which the failure rate increases with time, the failures tend to occur later, and there is a greater opportunity for a growing crack to be detected during a scheduled inspection.

5.2 EFFECT OF MODEL FOR INSPECTION PROBABILITY

The POD curves from the PNL piping inspection round robin are based on data for a single inspection of a specimen with a given flaw. In contrast, the current calculations address the detection of growing flaws for which several inspections occur periodically over the design life of the component. These calculations required assumptions to extrapolate the PNL curves for POD to the situation of multiple inspection.

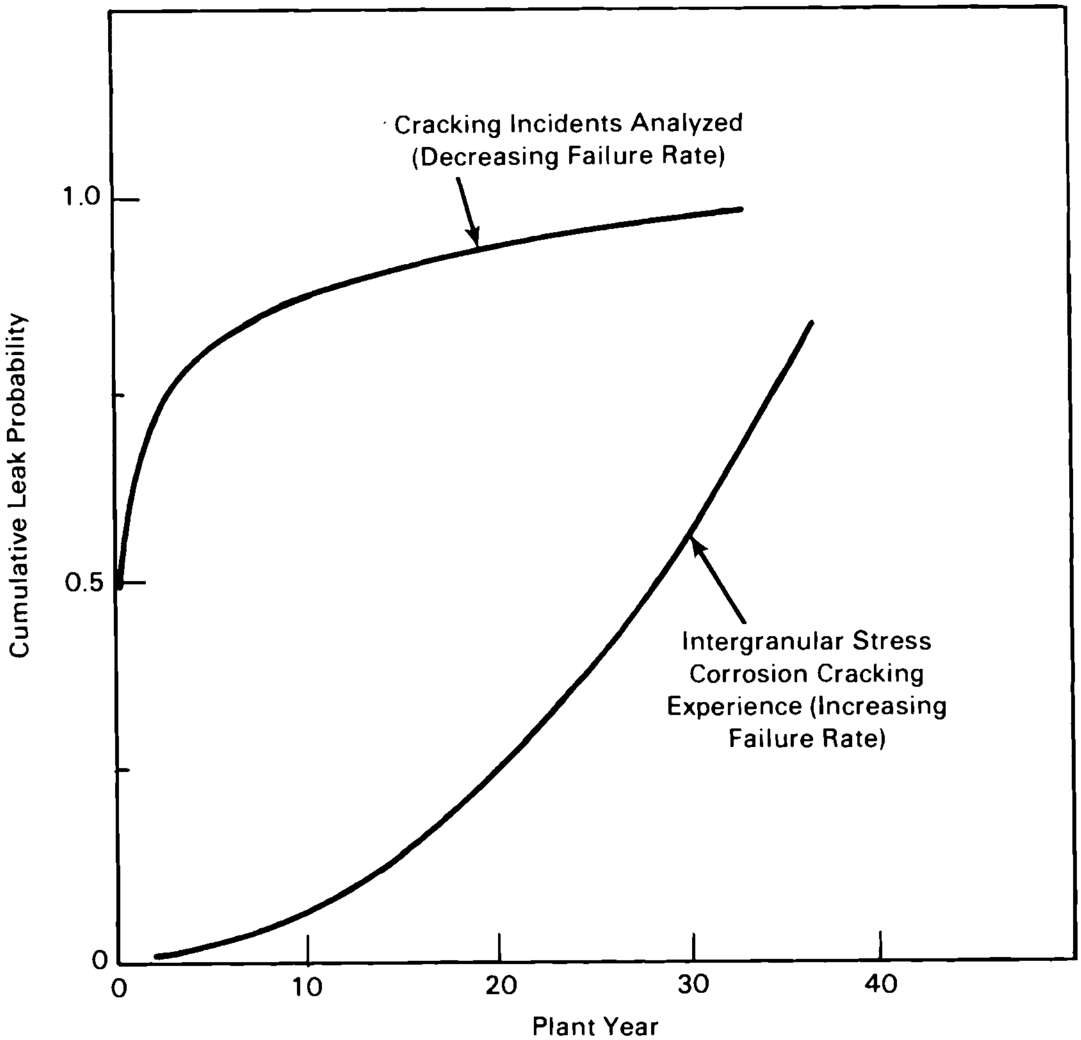

FIGURE 5.1. Comparison of Probability Trends

To describe crack detection for situations of repeated inspection, the following definitions are made.

- P_1 = probability of nondetection for the first inspection when flaw has depth a_1
- P_2 = probability of nondetection for the second inspection when flaw has depth a_2
- P_N = probability of nondetection for the N^{th} inspection when the flaw has depth a_N
- P = probability of nondetection over all N inspections.

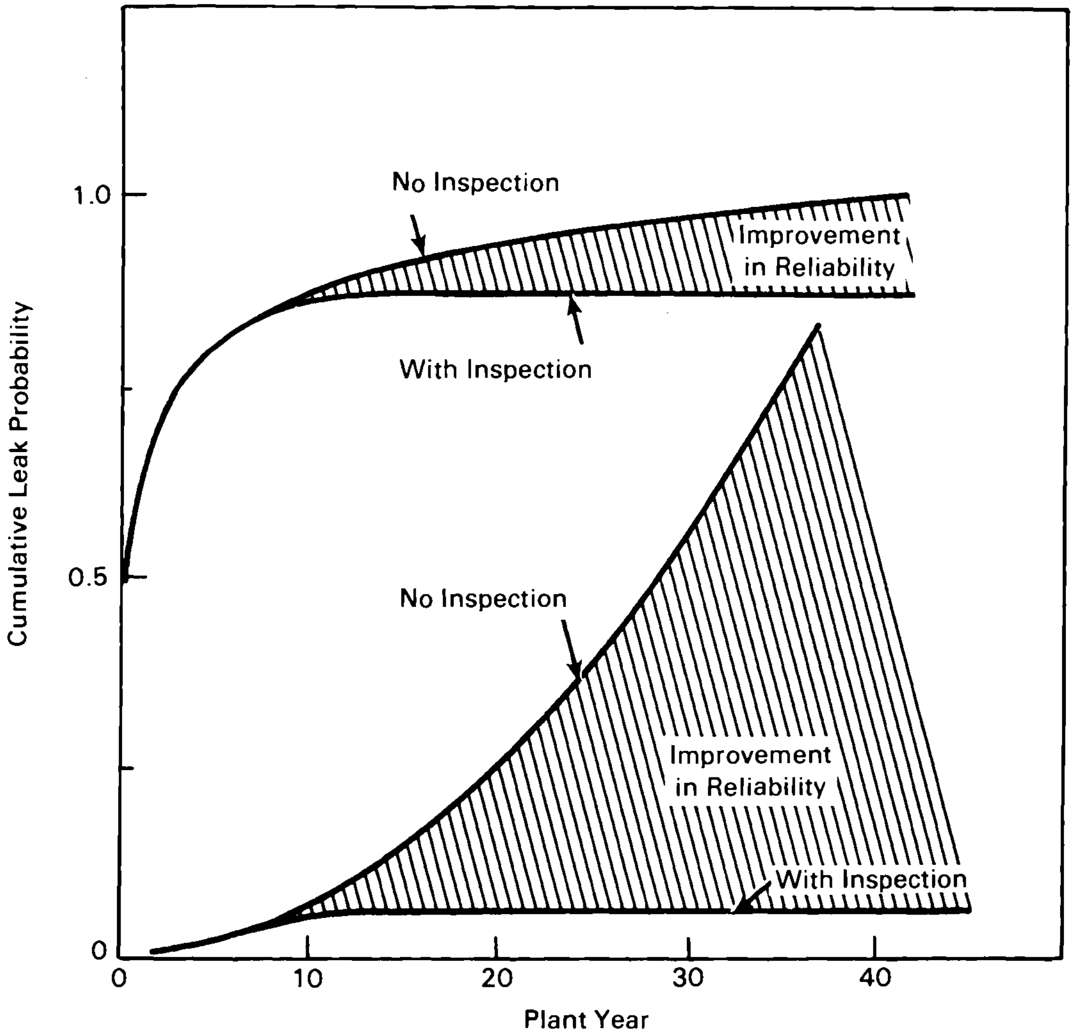

FIGURE 5.2. Impact of Inspection on Systems with Increasing Versus Decreasing Leak Probability

Two bounding assumptions can be made in the analysis of the multiple inspection scenario. These assumptions lead to quite different results for the predicted probability of detecting the growing flaw over the N inspections.

5.2.1 Bounding Assumption 1

It can be assumed that nondetection is due to purely random factors. Nondetection of a given crack in one inspection implies no systematic reason that this flaw will or will not be detected in a subsequent inspection. Under these conditions,

$$P = P_1 \cdot P_2 \ldots P_N \qquad (5.1)$$

This implies that a series of inspections with perhaps modest detection capability can produce a compounded detection capability much superior to any of the individual inspections. For example, consider N = 4 with the nondetection probabilities $P_1 = 0.50$, $P_2 = 0.20$, $P_3 = 0.10$, and $P_4 = 0.05$. The assumption of random factors implies that

$$P = P_1 \times P_2 \times P_3 \times P_4 = (0.50)(0.20)(0.10)(0.05) = 0.0005 \qquad (5.2)$$

Thus, the probability of not detecting the crack is only 5 chances in 10,000, which is a factor of 100 better than the best of the individual inspections (i.e., $P_4 = 0.05$). Calculations described in this resport are performed with the PRAISE code; the structure of this code gives detection probabilities that are governed by bounding assumption 1 as defined here.

5.2.2 Bounding Assumption 2

It can be assumed that nondetection is due to purely systematic factors such as crack tightness or signals from adjacent geometric reflectors. Nondetection of a given crack in one inspection implies that these same systematic factors will impede detection in subsequent inspections. Under these conditions it is conservative to assume that

$$P = P_N \qquad (5.3)$$

This means that the result of a series of inspections will give a detection capability governed by the detection capability for final inspection for which the flaw has its greatest depth and, hence, is most detectable. For the above example,

$$P = P_4 = 0.05 \qquad (5.4)$$

Thus, the crack detection capability is 100 times less effective than that predicted under assumption 1. The reader is referred to the Appendix for sample calculations based on assumption 2.

5.3 EFFECT OF PIPING RELIABILITY MODEL

Specific features of the piping reliability model used in the PRAISE code can affect the predicted impact of ISI on system reliability. The predicted shape of the failure probability curve (increasing versus decreasing failure rate) and the crack detection model have been reviewed in the above discussion. Other factors are now addressed.

It should be emphasized that the piping reliability model appears to correlate well with service experience for the two cracking incidents considered (feedwater line and recirculation line). In both cases the predicted failure probabilities have relatively high values for the times at which cracking was actually observed in service. However, data from the service experience are insufficient to determine if the analytical model correctly calculates that actual relationship between crack depth and time of reactor operation. This crack growth trend is an important factor that impacts the ability of NDE to detect a crack before it grows to an unacceptable depth.

The features of the analytical model that relate to the crack growth predictions are as follows:

- The cracks are assumed to initiate very early in the life of the component, so that crack growth rates rather than initiation times govern the time to failure. The introduction of the initiation time for cracks into the piping reliability model would tend to give more pessimistic predictions of the benefits of inspection.

- The growth of cracks is assumed to be adequately described by the simple Paris type relationship. Other, more complex, relationships could give crack depth versus time trends that either enhance or impede crack detection.

- It is assumed that no changes in reactor operating conditions occur. Thus, factors such as the level of stress, rate of cycling, and water chemistry remain unchanged. Changes in these factors would affect the crack depth versus time trends, and could either enhance or impede crack detection.

- Residual stresses have been neglected in the calculations of crack growth in welds. It is known that such stresses often have a significant effect on crack growth behavior, and may cause the crack growth process to slow down or arrest as the crack depth increases. Such an effect would increase the probability of crack detection.

- The model assumes that each critical location can fail only once during the reactor design life. As such, it is implied that a "perfect" repair occurs as soon as a crack is detected either through ISI or by detection of a leak associated with a through-wall crack.

These factors are listed to emphasize that conclusions regarding the benefits of ISI are, in many respects, dependent on the specific cracking incident, as well as on the specific model used to describe details of the cracking incident.

Figures 5.3 and 5.4 summarize the results of the calculations and indicate the relative impact of ISI period and NDE reliability (labeled "poor", "good", and "advanced"). Improvements in detection probability from poor to advanced teams are perhaps most clearly seen as a decrease in slope of the leak probability curves that occurs at the time of the first inspection. There is a clear difference among the performance levels for the three detection capabilities as measured by the slopes of the curves. However, for the selected scenarios, this decrease in slope (failure rate) has no impact on those components that develop leaks prior to this first inspection. In contrast, in systems with low leak probabilities early in life and with an associated increasing failure rate, the differences in slope will have a great impact on system reliability. The results for the case study described in the Appendix clearly show this positive impact.

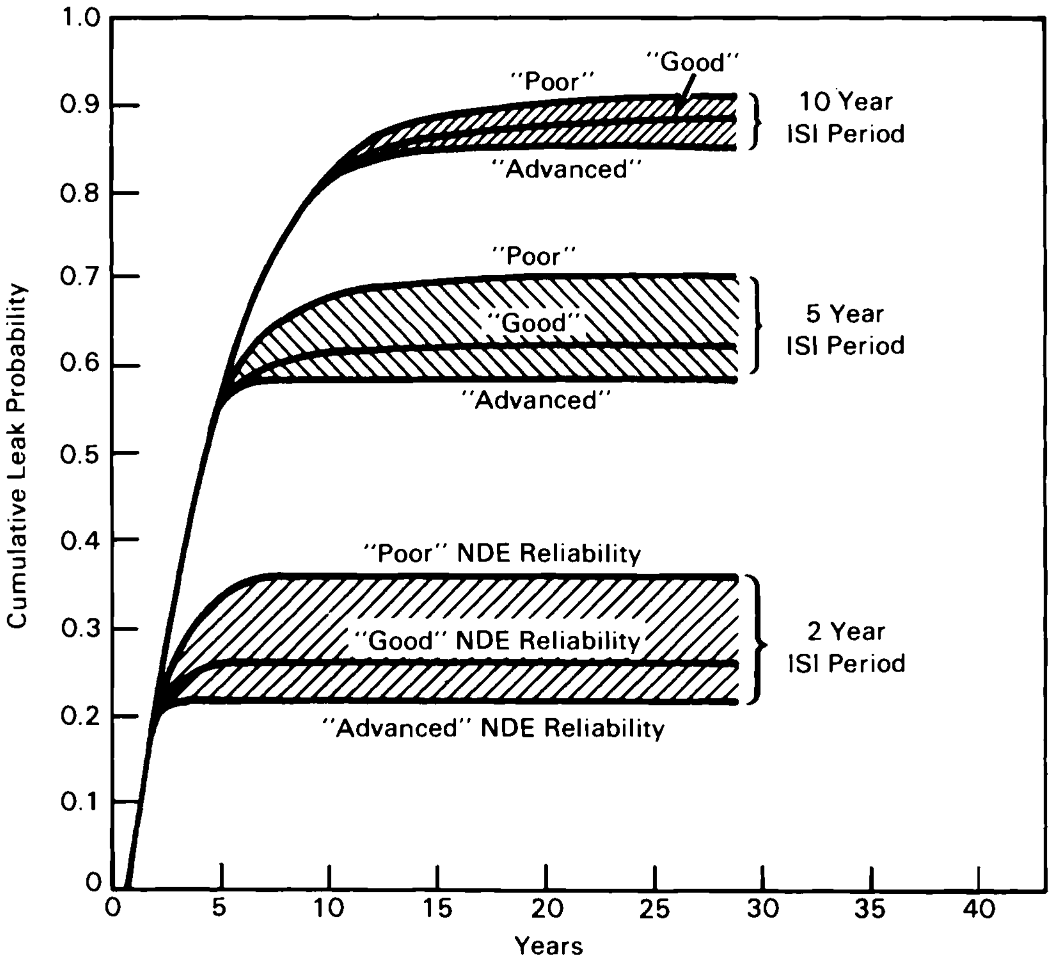

FIGURE 5.3. Thermal Fatigue of PWR Feedwater Line Nozzle

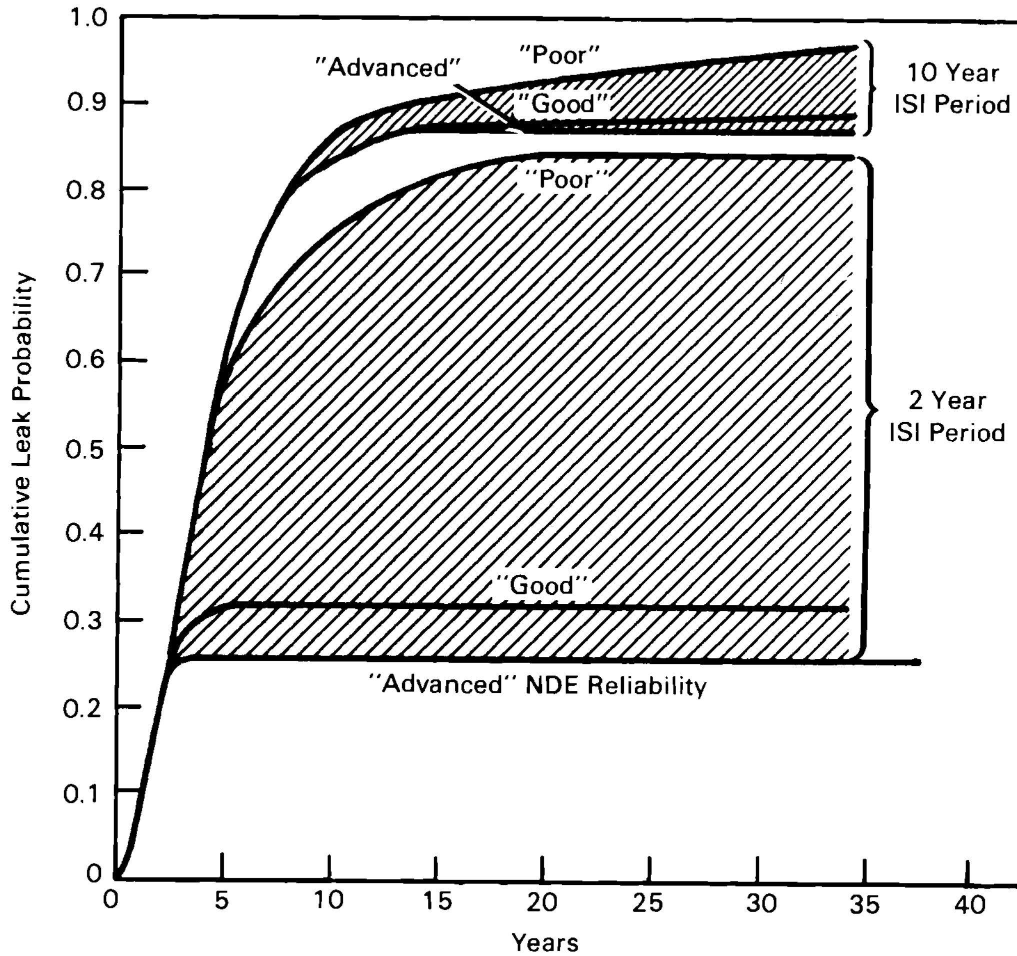

FIGURE 5.4. Intergranular Stress Corrosion Cracking of BWR Recirculation Line Nozzle Safe End

6.0 CONCLUSIONS

The impact of inservice inspection on the reliability of nuclear piping was evaluated for two service failure incidents using a probabilistic fracture mechanics approach. Based on the analysis results for the selected scenarios, four primary conclusions may be drawn:

- An effective inservice inspection requires a suitable combination of flaw detection capability and inspection schedule.
- An augmented inspection schedule is required for these specific lines with fast-growing flaws to ensure that the inspection is done before the flaws reach critical sizes.
- If flaws have the potential to grow to critical size in the early stage of plant operation, then the first service inspection is the most important one.
- For the PNL round robin study, the improvement as measured by reduction in leak probability from the "good" team to the "advanced" team is less than that from the "poor" team to the "good" team. This elimination of "poor" teams through training and qualification testing can produce significant benefits to ISI effectiveness.

Although the findings of this study may not be applicable to all piping systems, we believe that the probabilistic approach is suitable for assessing the impact of inservice inspection of other piping systems. It should be noted that the focus of this study was on lines with high failure probabilities early in the plant life. For these lines it appears that the effectiveness of inservice inspection is dominated by the inspection period rather than the flaw detection capability. For other lines that are not subject to high failure probabilities early in life, the effectiveness of inservice inspection (as shown in the Appendix) will be more strongly governed by the flaw detection capability.

REFERENCES

Bamford, W. H. 1980. "Technical Basis for Revised Reference Crack Growth Rate Curves for Pressure Boundary Steels in LWR Environment." ASME Journal of Pressure Vessel Technology. 102:433-442.

Bamford, W. H., A. Thurman and M. Mahlab. 1980. "Fatigue Crack Growth in Pressurized Water Reactor Feedwater Lines." Paper presented at the Joint Conference of the Pressure Vessels and Piping Materials, Nuclear Engineering and Solar Division, June 21-25, 1980, Denver, Colorado.

Boyce, L., and H. H. Woo. 1983. "Piping Reliability Analysis for Recirculation Safe Ends of a Boiling Water Reactor." Paper ml/5 presented at the 7th International Conference on Structural Mechanics in Reactor Technology, August 22-26, 1983, Chicago, Illinois.

Burghard, H. C., Jr. 1978. Metallurigical Investigation of Cracking in a Reactor Vessel Nozzle Safe-End. SWRI Project 02-5389-001, Southwest Research Institute, San Antonio, Texas.

Doctor, S. R., et al. 1983. "Integration of Nondestructive Examination Reliability and Fracture Mechanics." In Transactions of the Eleventh Water Reactor Safety Research Information Meeting. NUREG/CP-0047, U.S. Nuclear Regulatory Commission, Washington, D.C.

General Electric Company. 1982. The Growth and Stability of Stress Corrosion Cracks in Large-Diameter BWR Piping. EPRI-NP-2472, Electric Power Research Institute, Palo Alto, California.

Heasler, P. G., and F. A. Simonen. 1983. "Using Markov Chain Models to Describe Crack Growth and Inspection Effectiveness." ASME Paper No. 83-PVP-88 presented at the 4th National Congress on Pressure Vessel and Piping Technology, June 19-24, 1983, Portland, Oregon.

Mood, A. M., F. A. Graybill and D. C. Boes. 1974. Introduction to the Theory of Statistics. 3rd ed. McGraw-Hill, New York, New York.

NUREG-0531. Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants. February 1979.

NUREG-0691. Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors. September 1980.

NUREG/CR-1696. Integration of NDE Reliability and Fracture Mechanics. Pacific Northwest Laboratory, March 1981.

NUREG/CR-2189. Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant. Volume 5: Probabilistic Fracture Mechanics Analysis. Lawrence Livermore National Laboratory, August 1981.

NUREG/CR-2301. Fracture Mechanics Models Developed for Piping Reliability Assessment in Light Water Reactors. Lawrence Livermore National Laboratory, September 1981.

NUREG/CR-2716. Reactor Safety Research Programs - Quarterly Report, July - September, 1982. Pacific Northwest Laboratory, March 1983.

NUREG/CR-2801. Piping Reliability Model Validation and Potential Use for Licensing Regulation Development. Lawrence Livermore National Laboratory, January 1982.

NUREG/CR-3059. Parametric Calculations of Fatigue Crack Growth in Piping. Pacific Northwest Laboratory, March 1983.

NUREG/CR-3176. Crack Growth Evaluation for Small Cracks in Reactor Coolant Piping. Pacific Northwest Laboratory, April 1983.

Westinghouse Electric Corporation. 1980. Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants. WCAP-9693, Pittsburgh, Pennsylvania.

Woo, H. H., and C. K. Chou. 1982. Piping Reliability Analysis for Pressurized Water Reactor Feedwater Lines. UCRL-86216, Lawrence Livermore National Laboratory, Livermore, California.

APPENDIX

AN EVALUATION OF THE IMPACT OF INSERVICE INSPECTION ON STRESS CORROSION CRACKING IN BWR PIPING

This Appendix describes the results of probabilistic fracture calculations performed at PNL that evaluate the impact of ISI on intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) piping. The evaluation is for a different failure scenario and uses a fracture mechanics model that differs from that of the evaluations described in the body of the report.

The probability of crack detection curves (Figure A.1) from the PNL piping inspection round robin were applied to data on IGSCC. The IGSCC data were published by the Electric Power Research Institute (EPRI) and were based on operating experience with stainless steel piping in use at BWRs. These results from the EPRI/BWR Owners Group Research Project (General Electric Company 1982; Eason 1982) are indicated in Figures A.2 and A.3. The results of PNL's evaluation (shown in Figure A.4) indicate that a significant increase in reactor piping reliability can result from improved crack detection capability during NDE. Comparison of the results presented in this Appendix with results given in the body of this report show that the predicted benefits of ISI are sensitive to both the failure mode being addressed and the assumptions of the analytical model used to describe the inspection process.

The calculations utilized the crack growth curve of Figure A.2, which was extracted from Eason (1982). Although this curve is based on extensive analytical developments as well as laboratory experimental programs, it must be recognized that any prediction of the progress of IGSCC is subject to considerable uncertainty and is the topic of continuing research efforts. Nevertheless, the key feature of the particular curve in Figure A.2 is the prolonged period of shallow crack depth (starting from an initial depth of 0.010 in.) and the relatively slow rate of crack growth over most of the life of the component. Under these conditions of prolonged periods of small crack depth, the detection of cracks is difficult. If the probability of crack detection is to be high, inspections must occur relatively late in life, when the crack depth has increased to a relatively detectable size.

The curve of crack depth versus time as shown in Figure A.2 was not used in a direct or absolute sense. Rather, only the shape of the curve was utilized in these probabilistic calculations. The actual time to leak (or through-wall crack) was assumed to be described by the probabilistic curve shown in Figure A.3. This distribution from IE Bulletin No. 88-02 (NRC 1983) was empirically derived from an analysis of data from field experience with BWR piping systems. The curves in Figure A.3 show the fraction of welds that have required repair due to IGSCC cracking. This fraction increases with time as the days of reactor operation increase. The

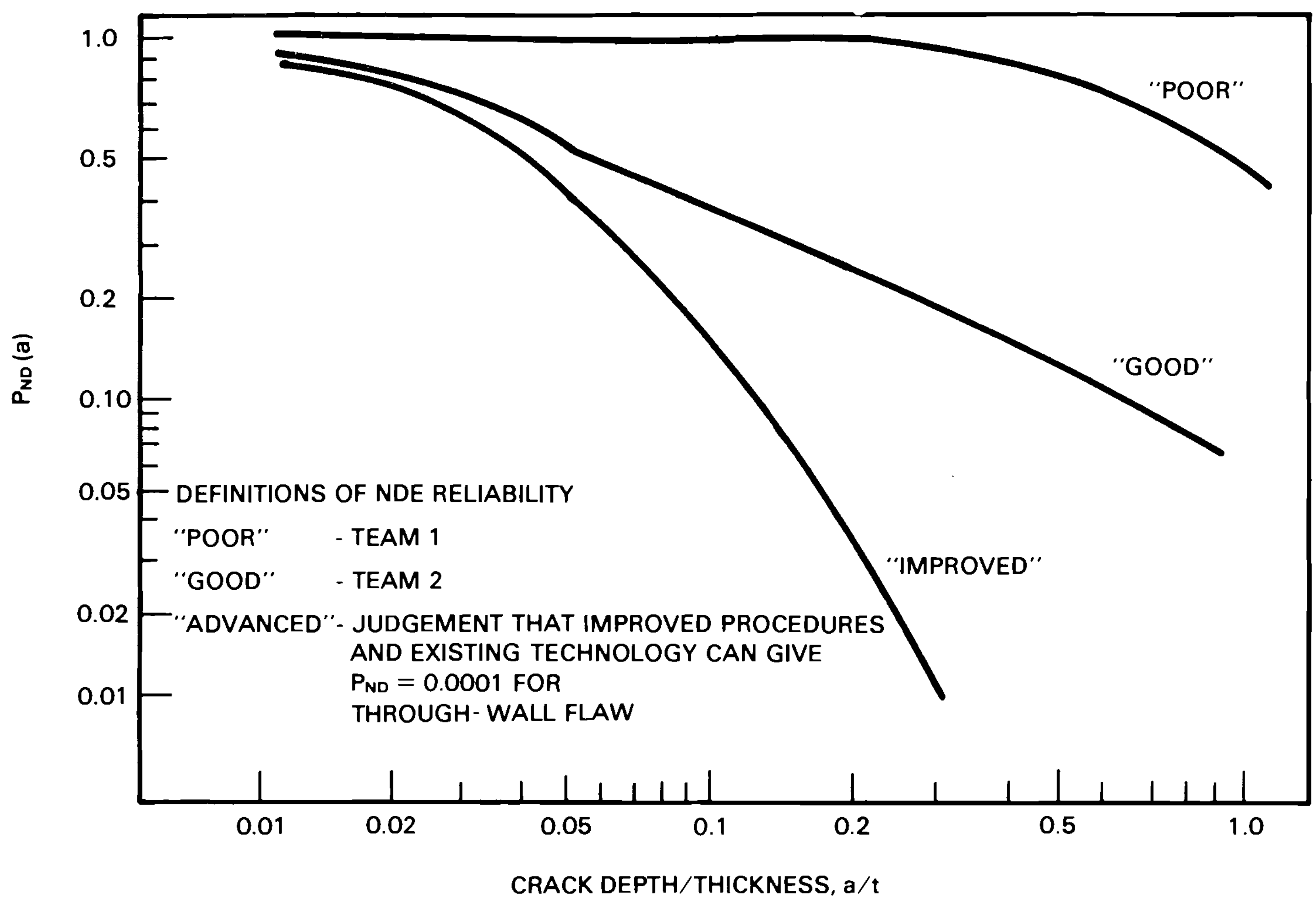

FIGURE A.1. Detection of Intergranular Stress Corrosion Cracking in 10-Inch Stainless Steel Pipe

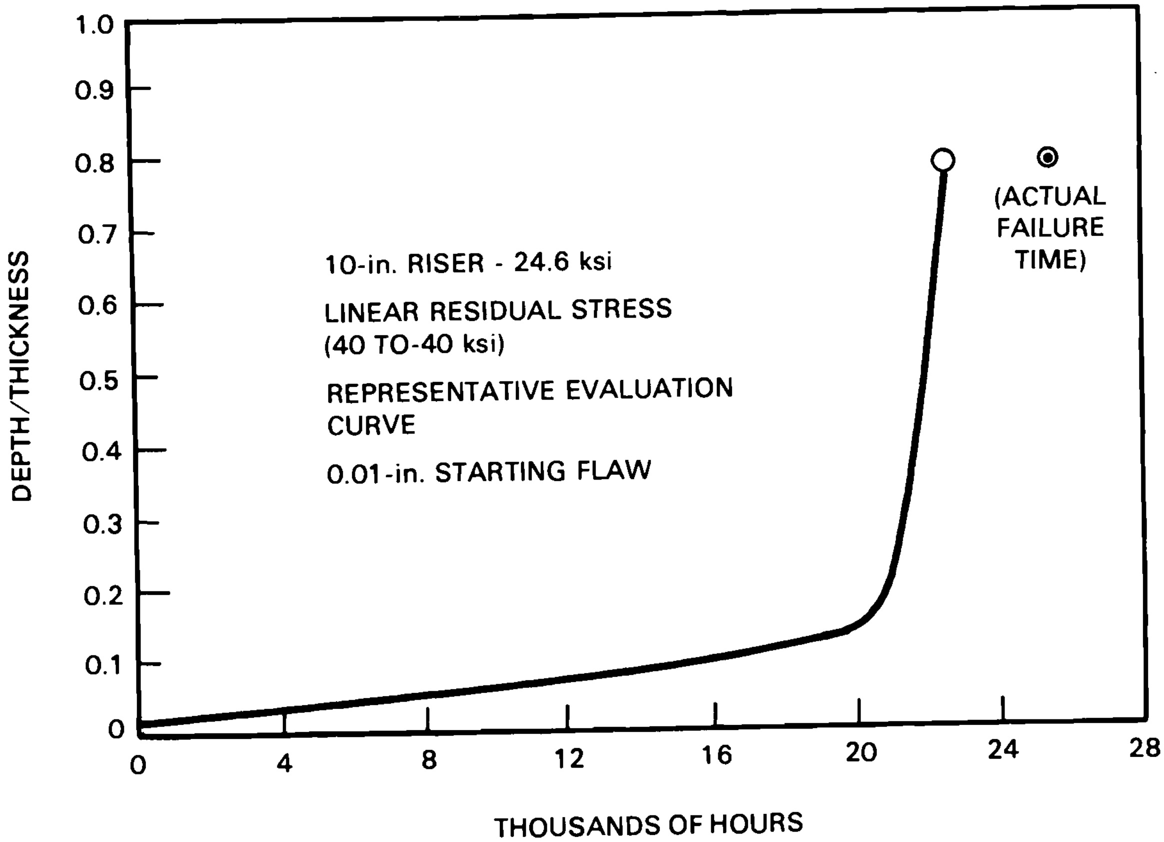

FIGURE A.2. Predicted Growth of Intergranular Stress Corrosion Crack from EPRI/BWR Owners' Group Program

Source: General Electric Company (1982).

analysis of data in IE Bulletin No. 88-02 showed that the data fell into two populations, designated as 1) recirculation bypass lines and 2) "other" BWR systems. As might be seen, there is considerable weld-to-weld variation in service life. This variation can be attributed to differences in such factors as welding residual stresses, coolant system chemistry, and degree of sensitization from welding, as well as other unspecified factors.

The probability of flaw nondetection curves of Figure A.1 are for "poor" and "good" NDE reliability and are representative of the range of performance by different teams in the PNL piping inspection round-robin. The "advanced" curve is an estimate of the level of performance that can be achieved within the limitations of field procedures and existing technology. For IGSCC there is clearly a significant range in team-to-team performance, although all teams including the "poor" team met minimum ASME Section XI Code requirements. The "poor" team detected only about 10 percent of flaws as deep as 40 percent of the wall thickness. In contrast, the "good" team detected about 80 percent of flaws of this depth. The "advanced" curve of Figure A.1 assigns a 10^{-4} probability of nondetection

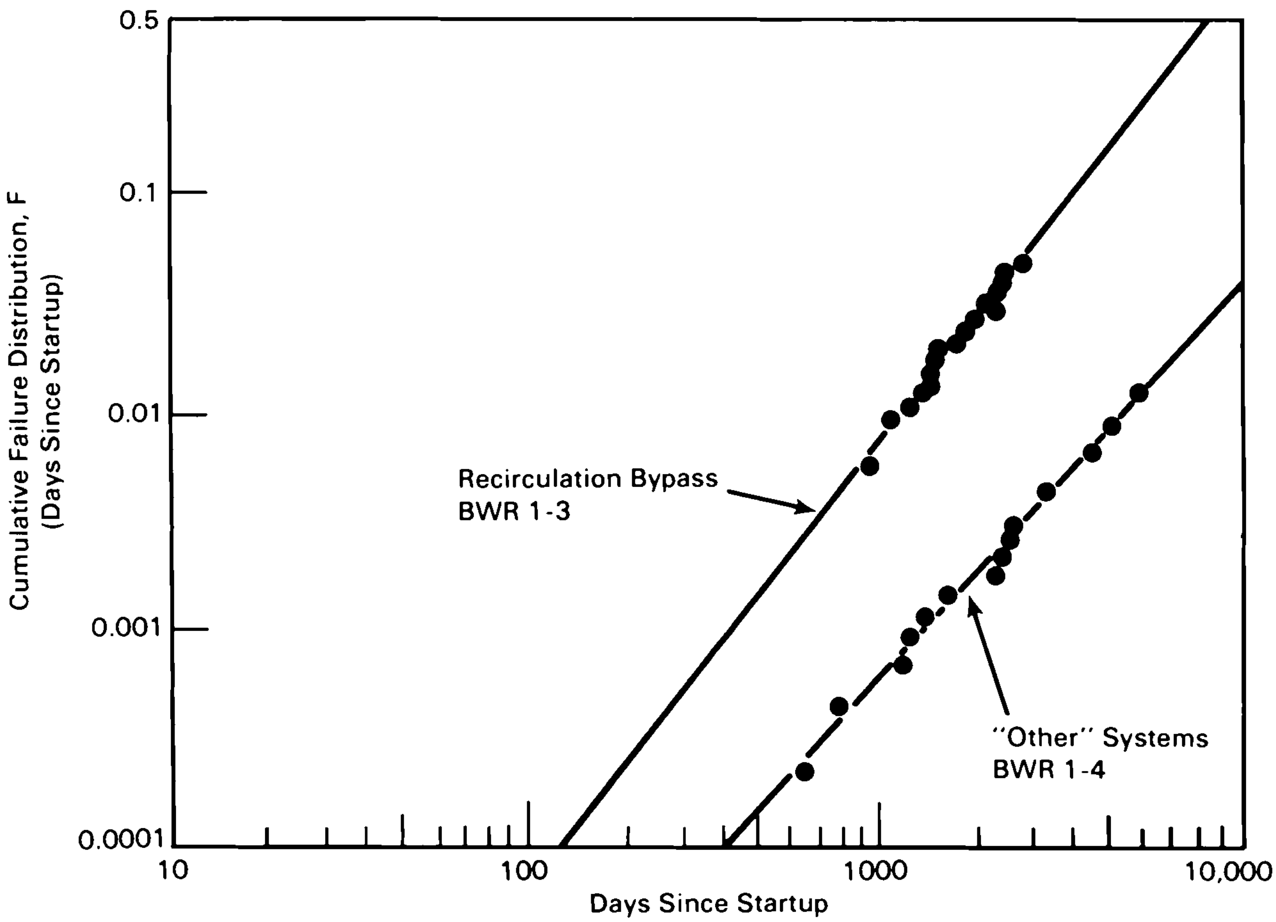

FIGURE A.3. Intergranular Stress Corrosion Cracking Experience in BWR Piping

Source: Eason (1982).

for a through-wall flaw. This assumes the use of advanced instrumentation, and assumes that human errors will be the major factor in nondetection of deep flaws. For small flaws, the instrumentation and physics of the detection process will be the major factors. A 90 percent probability of detection for a flaw with a depth of 10 percent of the wall was assumed.

In the fracture mechanics calculations it was necessary to apply the detection curves outside the parameters of the data from the piping inspection round robin. For pipe wall thickness other than the 10-inch Sch 80 pipe of the round robin, it was assumed that the detection probability was only a function of the ratio a/t, with the same function of a/t applying to all wall thicknesses. The round robin did not involve repeated inspection of actual growing IGSCC flaws.

For multiple inspections of a given growing crack, a lower bound on detection probability was estimated using methods from reliability theory. The detection probability was taken as the best obtained in any of the

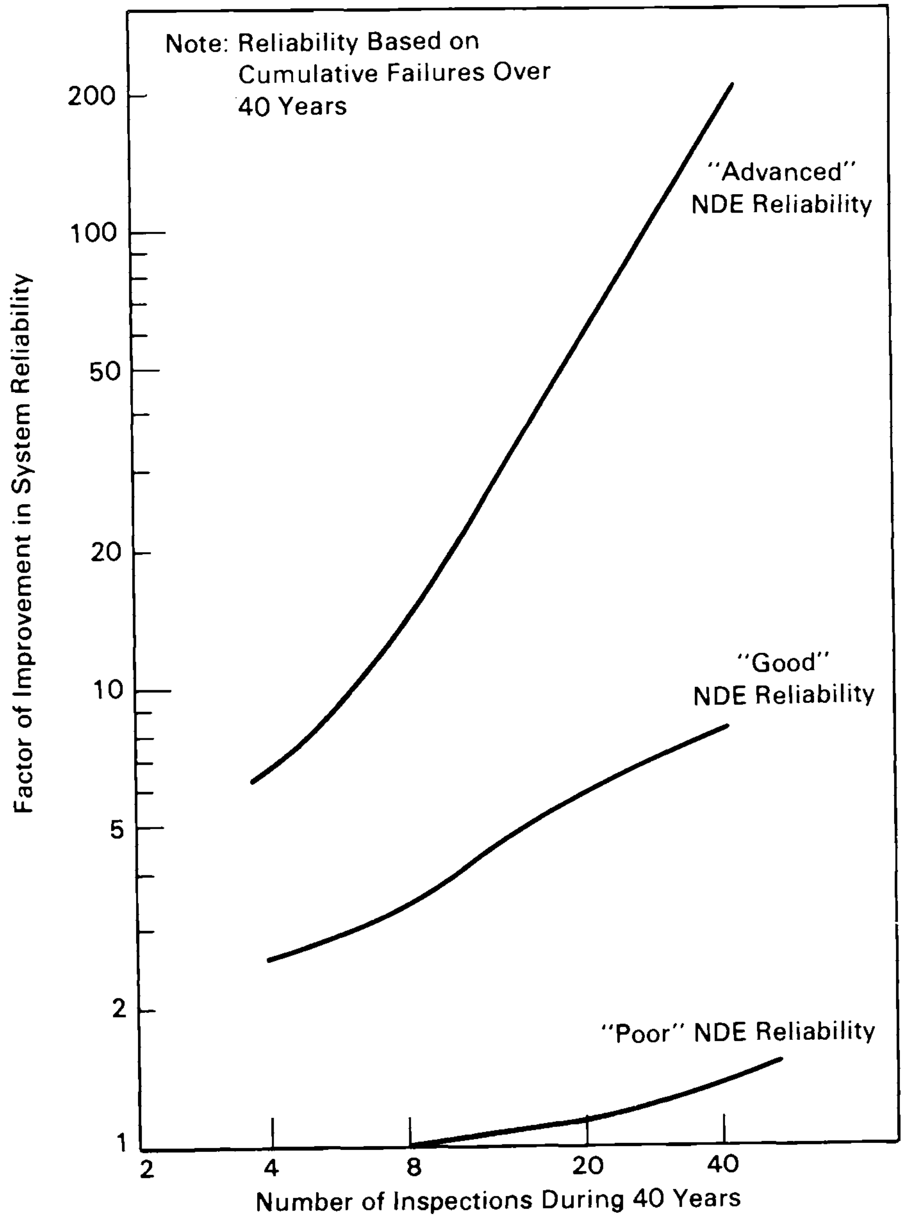

FIGURE A.4. Predicted Impact of NDE on the Occurrence of Leaks Caused by Intergranular Stress Corrosion Cracking

multiple inspections or, in effect, the detection probability corresponding to the last inspection for which the crack has its greatest depth. An alternative approach would have been to assume an optimistic upper bound behavior governed by the product of the probabilities of nondetection from the successive inspections. This approach was believed to be overly optimistic because it does not properly account for the systematic factors that may impede the detection of a given crack (e.g., crack tightness).

For the calculations underlying Figure A.4, the 40-year design life was divided into uniform inspection intervals, and the inspection was assumed to be performed at the midpoint of each interval. Inservice inspection intervals ranging from 10 years (ASME Code scheduled inspection) to 1 year were considered. In the probabilistic calculations a distribution of crack growth curves of the shape in Figure A.2 was simulated, with the time scale that defines end points of the curves having a distribution consistent with Figure A.3.

Figure A.4 shows the predicted impact of NDE on system reliability. The measure of reliability was defined as the number of leaks that occur over the 40-year design life of the plant. Factor of improvement is defined as the ratio

$$\text{Factor of improvement} = \frac{\text{Number of leaks without NDE}}{\text{Number of leaks with NDE}}$$

The results show the predicted improvement in system reliability that occurs as the number of inspections over the 40-year design life in increased.

It appears that "poor" NDE has little impact on system reliability, but "good" NDE can improve reliability by a factor of 2 to 10 depending on the ISI period. Even greater potential benefits can be achieved with "advanced" technology and procedures. Recent NRC requirements for enchanced NDE of BWR piping appear to have a positive impact on system reliability. In particular, the frequency of inspection required by NUREG-0313 and elimination of ineffective ISI via IEB 88-02 provides a factor of improvement of about 5 over code minimum ISI as provided by the "poor" NDE reliability curve of Figure A.1. The calculations of the impact of NDE on IGSCC support the following conclusions.

- There are clear and significant differences in the ability of alternative NDE procedures to detect IGSCC prior to leak.
- The "poor" team cannot approach a 50 percent probability of detecting IGSCC prior to leak, even with an extremely short inspection interval of once per year. Inspections of this quality are of essentially zero benefit, and are not justified on the basis of cost and radiation exposure concerns.
- The "good" team can readily a achieve a factor of improvement of 2 in preventing leaks due to IGSCC and can, with a short inspection interval, approach a factor of 10 improvement.
- "Advanced" ISI can readily achieve a factor of improvement of 10 in preventing IGSCC leaks; this can be increased to 100 with the inspection interval decreased sufficiently.

- Better inspection procedures ("good" versus "poor" and "advanced" versus "good") appear to offer a cost-effective option for enhancing piping performance. The results indicate that use of a better procedure can be more effective than a tenfold increase in the number of inspections with the continued use of an inferior procedure.

- A factor of 10 decrease in failure probability can be viewed as a reasonable estimate for the benefit that can be achieved through ISI.

- The results show that relatively short inspection intervals (e.g., one inspection per year of operation) are required to prevent leaks due to IGSCC in failure-prone welds. Such inspection intervals may be practical but only as a temporary measure for a few selected welds. However, such intervals would be impractical as a plant-wide measure to assure piping integrity.

REFERENCES

Eason, E. D. 1982. "Stress Corrosion Cracking in Boiling Water Reactor (BWR) Piping." In Proceedings of the Workshop on Nuclear Power Plant Aging, NUREG/CP-0036, U.S. Nuclear Regulatory Commission, Washington, D.C.

General Electric Company. 1982. The Growth and Stability of Stress Corrosion Cracks in Large Diameter BWR Piping. EPRI-NP-2472, Electric Power Research Institute, Palo Alto, California.

NUREG-0313 (Rev. 1). Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping. 1972.

U.S. Nuclear Regulatory Commission. 1983. Stress Corrosion Cracking in Large-Diameter Stainless Piping at BWR Plants. IE Bulletin No. 88-02, U.S. Nuclear Regulatory Commission, Washington, D.C.

DISTRIBUTION

No. of Copies

OFFSITE

U.S. Nuclear Regulatory Commission
Division of Technical Information and Document Control
7920 Norfolk Avenue
Bethesda, MD 20014

5 J. Muscara
Materials Engineering Branch
Engineering Technology Division
Nuclear Regulatory Commission
Mail Stop 5650NL
Washington, D.C. 20555

B. D. Liaw
Materials Engineering Branch
Division of Engineering
Nuclear Regulatory Commission
Mail Stop 318
Washington, D.C. 20555

W. S. Hazelton
Materials Engineering Branch
Division of Engineering
Nuclear Regulatory Commission
Mail Stop 318
Washington, D.C. 20555

M. R. Hum
Materials Engineering Branch
Division of Engineering
Nuclear Regulatory Commission
Mail Stop 318
Washington, D.C. 20555

J. R. Quinn
Electric Power Research Institute
3412 Hillview
Palo Alto, CA 94303

No. of Copies

J. Stronsnider, Jr.
U.S. Nuclear Regulatory Commission
Division of Engineering Technology/RES
Materials Engineering Branch
Washington, D.C. 20555

R. W. Klecker
U.S. Nuclear Regulatory Commission
Division of Engineering Branch
Washington, D.C. 20555

W. J. Collins
Office of Inspection and Enforcement
Nuclear Regulatory Commission
Washington, D.C. 20555

R. E. Johnson
U.S. Nuclear Regulatory Commission
Generic Issues Branch
Mail Stop 268 Phillips Bldg.
Washington, D.C. 20555

G. Weidenhamer
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Mail Stop 1130-SS
Washington, D.C. 20555

M. Vagins
U.S. Nuclear Regulatory Commission
Division of Engineering Technology/RES
Materials Engineering Branch
Washington, D.C. 20555

W. J. Shack
Argonne National Laboratory
9700 Cass Ave.
Argonne, IL 60439

J. M. Bloom
Babcock & Wilcox Company
Alliance Research Center
P.O. Box 835
Alliance, OH 44601

W. A. Van Der Sluys
Babcock & Wilcox Company
Alliance Research Center
1562 Beeson Street
Alliance, OH 44601

G. M. Wilkowski
Battelle Columbus Laboratiories
505 King Ave.
Columbus, OH 43201

P. C. Paris
c/o Fracture Proof Design Corp.
77 Maryland Plaza
St. Louis, MO 63108

T. J. Griesbach
Electric Power Research Institute
Nuclear Safety Analysis Center
3412 Hillview Avenue
Palo Alto, CA 94303

D. M. Norris
Electric Power Research Institute
Nuclear Systems and Materials Department
3412 Hillview Avenue
Palo Alto, CA 94303

R. C. Cipolla
Aptech Engineering Services
795 San Antonio Road
Palo Alto, CA 94303

T. U. Marston
Electric Power Research Institute
Nuclear Systems and Materials Dept.
3412 Hillview Avenue
Palo Alto, CA 94303

C. E. Buchalet
Framatome
Tour Fiat/Cedex 16
92-084 Paris La Defense
FRANCE

L. J. Chockie
General Electric Company
Nuclear Energy Business Operation
175 Curtner Avenue, M/C 363
San Jose, CA 95125

S. Ranganath
General Electric Company
175 Curtner Avenue, M/C 777
San Jose, CA 95125

S. Yukawa
General Electric Company
Turbine Technology Laboratory
One River Road, Bldg. 55, Rm 113
Schenectady, NY 12345

F. L. Becker
J.A. Jones Applied Research Co.
P.O. Box 217097
Charlotte, NC 28221

C. K. Chou
Lawrence Livermore National Lab
P.O. Box 808
Livermore, CA 94550

H. H. Woo
Lawrence Livermore National Lab
P.O. Box 808
Livermore, CA 94550

M. E. Mayfield
Materials Engineering Associates
9700 B Palmer Highway
Lanham, MD 20706

E. Debarba
Northeast Utilities Service Co.
P.O. Box 270
Hartford, CT 06101

M. Kupinski
Northeast Utilities Service Co.
P.O. Box 270
Hartford, CT 06101

P. C. Riccardella
Structural Integrity Associates
3150 Almaden Expressway
Suite 226
San Jose, CA 95118

J. G. Merkle
Oak Ridge National Laboratory
P.O. Box Y
Oak Ridge, TN 37830

D. O. Harris
Failure Analysis Associates
2225 East Bayshore
Palo Alto, CA 94303

W. H. Bamford
Westinghouse Electric Corp.
P.O. Box 355
Pittsburgh, PA 15230

No. of
Copies

ONSITE

50 Pacific Northwest Laboratory

M. C. C. Bampton
C. A. Bennett (HARC)
S. H. Bush
S. L. Crawford
S. R. Doctor (3)
H. R. Hartzog
P. G. Heasler
K. I. Johnson
A. M. Liebetrau
G. A. Mart
M. R. Mullen
L. T. Pedersen
P. J. Pelto
G. J. Posakony
R. E. Rhoades
E. P. Simonen
F. A. Simonen (20)
J. C. Spanner
A. M. Sutey
T. T. Taylor (3)
L. D. Williams
Technical Information (5)
Publishing Coordination (2)

NRC FORM 335 (11-81) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)	NUREG/CR-3869 PNL-5149
4. TITLE AND SUBTITLE (Add Volume No., if appropriate)	Analyses of the Impact of Inservice Inspection Using a Piping Reliability Model
2. (Leave blank)	
3. RECIPIENT'S ACCESSION NO.	
7. AUTHOR(S)	F. A. Simonen, H. H. Woo
5. DATE REPORT COMPLETED	MONTH: July; YEAR: 1984
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Pacific Northwest Laboratory P.O. Box 999 Richland, WA 99352
DATE REPORT ISSUED	MONTH: August; YEAR: 1984
6. (Leave blank)	
8. (Leave blank)	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)	Division of Engineering Technology Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission Washington, DC 20555
10. PROJECT/TASK/WORK UNIT NO.	
11. FIN NO.	FIN B2289
13. TYPE OF REPORT	Topical Report
PERIOD COVERED (Inclusive dates)	
15. SUPPLEMENTARY NOTES	
14. (Leave blank)	

16. ABSTRACT (200 words or less)

This report presents the results of a study of the impact of inservice (ISI) programs on the reliability of specific nuclear piping systems that have actually failed in service. Two major factors are considered in the ISI programs: one is the capability of detecting flaws; the other is the frequency of performing ISI. A probabilistic fracture mechanics model issued to estimate the reliability of two nuclear piping lines over the plant life as functions of the ISI programs. Examples chosen for the study are the PWR feedwater steam generator nozzle cracking incident and the BWR recirculation reactor vessel nozzle safe-end cracking incident. The results show that an effective inservice inspection requires a suitable combination of flaw detection capability and inspection schedule. An augmented inspection schedule is required for piping with fast-growing flaws to ensure that the inspection is done before the flaws reach critical sizes. Also, the elimination of "poor" inspection teams through training and qualification testing can produce significant benefits to ISI effectiveness.

17. KEY WORDS AND DOCUMENT ANALYSIS	inservice inspection programs (ISI)
17a. DESCRIPTORS	
17b. IDENTIFIERS/OPEN-ENDED TERMS	
18. AVAILABILITY STATEMENT	Unlimited
19. SECURITY CLASS (This report)	Unclassified
20. SECURITY CLASS (This page)	Unclassified
21. NO. OF PAGES	
22. PRICE	$