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Life Prediction and Monitoring of Nuclear Power Plant Components for Service-Related Degradation

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This paper describes industry programs to manage structural degradation and to justify continued operation of nuclear components when unexpected degradation has been encountered due to design materials and/or operational problems. Other issues have been related to operation of components beyond their original design life in cases where there is no evidence of fatigue crack initiation or other forms of structural degradation. Data from plant operating experience have been applied in combination with inservice inspections and degradation management programs to ensure that the degradation mechanisms do not adversely impact plant safety. Probabilistic fracture mechanics calculations are presented to demonstrate how component failure probabilities can be managed through augmented inservice inspection programs. [DOI: 10.1115/1.1344237]

Introduction

Evaluations of nuclear power plant safety have assumed that passive components such as pressure vessels and piping systems have very low failure probabilities, such that failures of these components make only negligible contributions to plant risk (e.g., core damage frequency). In the U.S. and other countries the design, fabrication, inspection, and maintenance of piping and vessels have followed the conservative engineering practices specified by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Codes. The relatively small number of significant failures that have occurred during operating experience has demonstrated the soundness of the ASME code procedures. However, many plants will be approaching their design lives (e.g., 40 yr), with the expectation that continued operation beyond the original design period will need to be justified. Therefore, the assumption of continued high levels of structural reliability requires an extrapolation beyond the current base of operating experience that must be addressed as part of plant life extension efforts.

Whereas the replacement of active components (mechanical and electrical) is a routine part of plant maintenance, large-scale replacements of vessel and piping components is not economically feasible. The challenge is to make realistic life predictions, and to establish a high level of confidence in these predictions. A desired objective is to ensure that passive components continue to make only negligible contributions to plant risk relative to less easily managed contributions to risk such as failures of active components and operator errors.

Fatigue damage was originally identified as the life-limiting degradation mechanism for many pressure vessel and piping components during the design of nuclear power plants. With an aging population of operating plants, certain structural locations may exceed their original design lives based on calculated values of fatigue usage factors, although there has been no evidence of degradation as the predicted fatigue lives have been approached or exceeded. On the other hand, various degradation mechanisms, such as thermal fatigue, environmentally assisted fatigue, stress corrosion cracking, and flow-accelerated corrosion, were not anticipated during design, and have resulted in actual structural failures and early replacements and repairs to components.

This paper describes efforts in the nuclear industry to justify continued operation, with particular attention to components that have exhibited degradation or which may exceed original limits based on predicted design lives. Two technical bases for continued operation are presented. The first approach makes use of knowledge gained from plant operating experience to identify and manage degradation mechanisms. These mechanisms may not have been anticipated during the design of the plant, but given their actual occurrence have the potential to cause failures by small leaks, large leaks, or ruptures. The second approach addresses failure mechanisms, such as fatigue due to anticipated plant operating transients, which design calculations show the potential for occurrence, but for which plant operating experience has not yet shown any evidence of actual occurrence. Probabilistic fracture mechanics calculations demonstrate that an augmented level of inservice inspection can ensure acceptable failure probabilities for fatigue critical components.

Management Programs for Service-Related Degradation

Studies by Bush [1,2], Jamali [3], Thomas [4], and Wright et al. [5] have shown that piping failures are generally due to operational conditions, materials selection, and design features that were not adequately addressed or perhaps not addressed at all in the design of plant systems. On the other hand, those mechanisms such as mechanical fatigue due to anticipated operational transients, which have been considered as part of the plant design, have been addressed in a very effective manner and are seldom (if ever) the cause of service related failures.

Given the large number of potential service-related degradation mechanisms, the nuclear industry has adopted monitoring and managing practices, rather than life prediction and retirement practices, to ensure safe and reliable systems. The strategy involves the following steps:

- a reporting system to ensure that the industry can respond to adverse operating experience (detecting of cracking or leakage) before unanticipated degradation mechanisms impact a large number of plants and/or result in safety significant structural failures;
- augmented inservice inspections that are targeted to specific systems, materials, and/or operating conditions to ensure detection of early stages (small cracks or minimal wall thinning) of degradation mechanisms;

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• changes to plant operating conditions (e.g., improved water chemistries) to decrease degradation rates to negligible levels;
• replacement of inadequate piping and vessel components with improved materials and/or design practices.

Some industry programs have been effective in responding to both anticipated and unanticipated degradation mechanisms. Ongoing efforts by the nuclear power industry to address service-related problems are described in the forthcoming.

ASME Section XI Inservice Inspection

Formal integrity management programs were first established for nuclear power plants in the early 1970s. Until that time, limited attention was given to the needs of inservice inspections (ISI) in early nuclear power plant designs. It was generally believed that system radioactivity would render periodic inspections impractical. Since the nuclear plant systems were being designed and constructed to higher quality standards than those applied to fossil plants, ISI was assumed to be unnecessary. However, by the late 1960s, the number of service induced defects requiring the repair of nuclear system components increased. This prompted a cooperative effort between the U.S. Atomic Energy Commission (AEC) and industry to develop inspection program standards under the oversight of the American National Standards Institute (ANSI) and the American Society of Mechanical Engineers (ASME). By 1970, the ASME Boiler and Pressure Vessel Code, Section XI "Inservice Inspection of Nuclear Reactor Coolant Systems" was published.

Over 50 percent of the inspection categories pertained to welds. The inspection locations were primarily selected based on factors such as: component design stresses, estimated fatigue usage, dissimilar metal welds, and irradiation effects.

Originally, service-induced flaws were assumed to occur from random causes, at random locations, and at random times. Therefore, the Section XI inspection program relied upon a representative sampling of weld locations and randomized the timing of inspections as much as possible. The examination procedures and flaw acceptance standards assumed that the principle cause of failure would be due to fatigue stress cycles created by anticipated design cyclic loads (i.e., thermal fatigue). For Class 3 systems (i.e., service water systems) Section XI program requirements are limited to periodic leak and hydrostatic pressure testing—no volumetric or surface examinations are required.

Service Experience Insights

Service experience [6,7] has shown no correlation between actual failure probability and design stresses in the Design Report. Failures (cracks, leaks, and breaks) typically result from degradation mechanisms and loading conditions (i.e., IGSCC, flow accelerated corrosion, thermal stratification, etc.) not anticipated in the original design. Depending on the degradation mechanism present, failures are not necessarily limited to weld locations.

The Swedish Nuclear Power Inspectorate (SKI) compiled a database on reported piping failure events (leaks, breaks, and ruptures) in U.S. commercial nuclear power plants [8]. This database includes a total of 1511 piping and piping component failures on various safety and balance-of-plant (BOP) systems that have been reported to U.S. regulatory bodies from December 1961 through October 1995, encompassing 2068 reactor operating years. Figure 1 shows the distribution of all piping failures according to the following causes:

• corrosion fatigue—CF
• thermal fatigue—TF
• stress corrosion cracking—SCC
• corrosion attack—COR
• erosion and cavitation—E-C
• flow-accelerated corrosion (i.e., erosion corrosion)—E/C
• high-cycle vibration fatigue—VF
• water hammer—WH

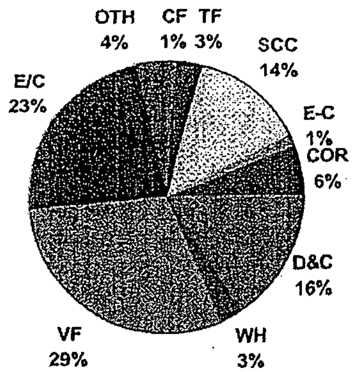

Fig. 1 Piping failure events in U.S. nuclear plants (1961–1996)

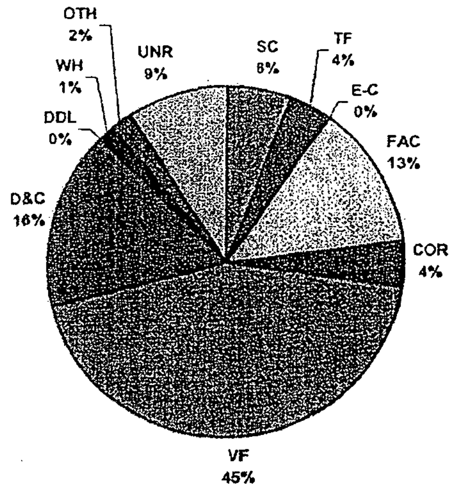

Fig. 2 Service failures in small-bore piping (<2 in. NPS)

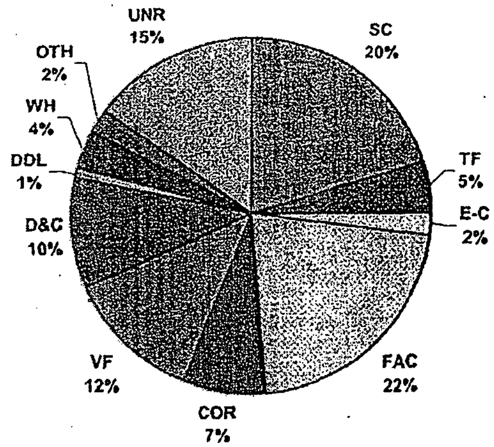

Fig. 3 Service failures in large-bore piping (>2 in. NPS)

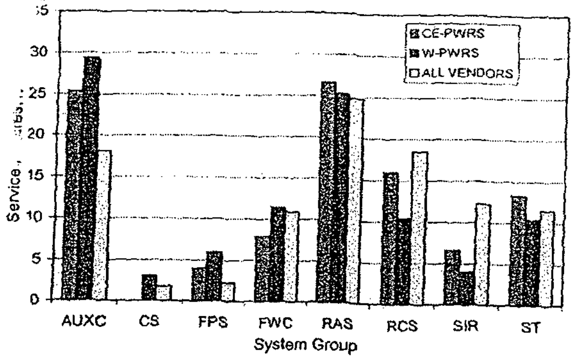

Fig. 4 Service failures by system groups

- design and construction errors—D&C
- other—OTH

The data of Fig. 1 shows that only 3 percent of all the reported service-induced piping failures were caused by thermal fatigue. This suggests that in their present form the ASME Section XI ISI program requirements are relatively ineffectual with regard to reducing overall piping failure probabilities. Approximately 72 percent of all reported failures were due to degradation mechanisms not addressed by ASME Section XI. For approximately 25 percent of all the reported events, piping failure resulted from failure mechanisms that were not associated with a particular damage mechanism. These include pipe failures caused by transient loading conditions and other factors such as construction errors, water hammer, overpressure, and frozen pipes. In these cases traditional inspection programs may be ineffective in preventing or reducing the piping failure probability.

Figures 2 and 3 compare service failures in small bore (<2 in. NPS) and larger bore (>2 in. NPS) piping. Approximately three-quarters of the reported service failures in small-bore piping were caused by either high-cycle vibration fatigue (VF), flow-accelerated corrosion (FAC), or design and construction errors (D&C). Almost half (45 percent) of the small-bore pipe failures were due to vibration fatigue. The majority of these failures occurred at socket-welded connections in poorly supported or cantilevered vent and drain lines <1 in. NPS.

Over 50 percent of the reported large bore piping service failures were caused by stress corrosion cracking (SCC), VF, and FAC. SCC and FAC accounted for 42 percent, and VF accounted for 12 percent of the reported failures. Sixteen percent of the small-bore failures were caused by D&C compared to 10 percent for large-bore piping. This appears to reflect field welding and fabrication difficulties associated with smaller-diameter piping.

Figure 4 shows the number of service failures reported in several plant system groups. Each system group is described in Table 1. System group service experience for Combustion Engineering PWRs, for Westinghouse PWRs, and for ALL plants is shown. Over half of the reported service failures in Combustion Engineering and Westinghouse PWRs occurred in reactor auxiliary systems (component cooling water, chemical volume and control, spent fuel pool cooling, radwaste, etc.) and auxiliary cooling systems (service water, salt water cooling, main circulating water, etc.).

Table 1 Service failure data system grouping

GROUP DESIGNATOR	SYSTEM GROUP DESCRIPTION	REPRESENTATIVE SYSTEM NAMES
RCS	Reactor Coolant System	Pressurizer, Reactor Coolant System
SIR	Safety Injection and Recirculation System	High and Low Pressure Safety Injection, Residual Heat Removal, Shut Down Cooling, Accumulator or other passive injection systems
CS	Containment Spray System	Containment Spray System
RAS	Reactor Auxiliary Systems	Component Cooling Water, Chemical Volume and Control, Spent Fuel Pool Cooling, Radwaste (no salt or dirty water systems)
AUXC	Auxiliary Cooling Systems	Service Water, Salt Water Cooling, Main Circulating Water, and other dirty water systems
FWC	Feedwater and Condensate Systems	Main Feedwater System, Auxiliary Feedwater System, Condensate System
ST	Main and Auxiliary Steam Systems	Main and Auxiliary Steam Systems
	Fire Protection Systems	Fire Protection System

Augmented Inspection Programs

For some of the more significant causes of piping failures, augmented inspection programs have been implemented. These programs, many of which have been mandated by the NRC, are designed to address component integrity relative to the impacts associated with a specific damage mechanism.

Intergranular Stress Corrosion Cracking. Stress-corrosion cracking (SCC) refers to cracking caused by the simultaneous presence of tensile stress and a corrosive medium. The important variables affecting SCC are temperature, water chemistry, metal composition, stress, and metal microstructure. Both intergranular (cracking proceeds along the material grain boundary) and transgranular (crack growth is not affected by the presence of grain boundaries) cracking have been observed. Intergranular stress corrosion cracking (IGSCC) results from a combination of sensitized materials (caused by a depletion of chromium in regions adjacent to the grain boundaries in weld heat-effected zones), high stress (residual welding stresses), and a corrosive environment (high level of oxygen or other contaminants).

IGSCC is encountered most frequently in austenitic stainless steels that become sensitized through the welding process and are subjected to BWR operating environments. The susceptible areas extend into the base material a few millimeters beyond either side of the weld—the weld "heat-affected zone." Welds in materials considered to be resistant to sensitization from welding are not susceptible to degradation from IGSCC.

A discussion of the IGSCC problems in BWR nuclear plants and the associated augmented program requirements can be found in Generic Letter 88-01 [9] and in NUREG 0313 [10]. The industry was required to establish programs that included the following:

- implement piping replacements or other measures to mitigate IGSCC;
- augment the existing Section XI ISI program to incorporate an inspection scope and frequency consistent with the extent of mitigation actions implemented;
- improve leak detection and monitoring programs;
- implement programs to improve NDE inspector performance in the detection and characterization of IGSCC damage.

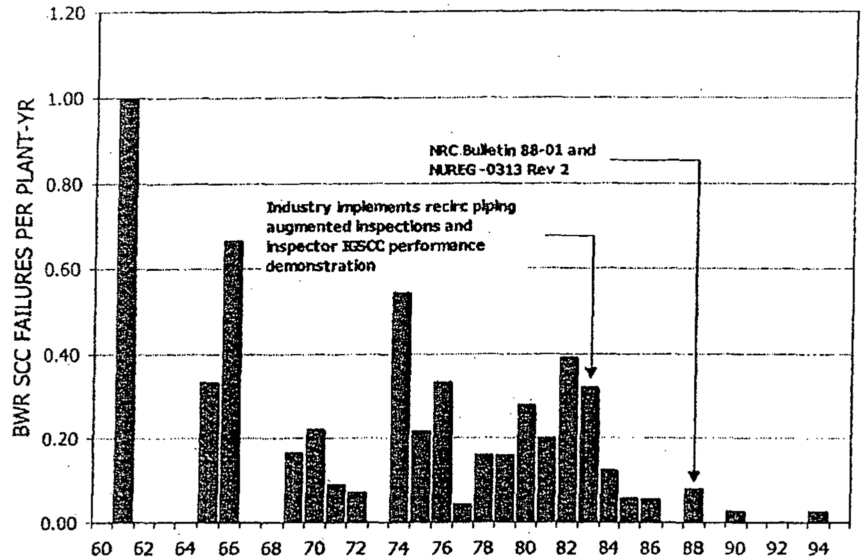

Fig. 5 BWR SCC failures per plant year

Figure 5 shows that since the implementation of these program requirements, the frequency of IGSCC caused piping failures, which might otherwise have increased, has instead been significantly reduced.

Flow-Accelerated Corrosion. Flow-accelerated corrosion (FAC) is a complex phenomenon that exhibits attributes of erosion and corrosion in combination. Factors that influence whether FAC is an issue are velocity, dissolved oxygen, pH, moisture content of steam, and material chromium content. Carbon steel piping with chromium content greater than 1 percent and austenitic steel piping is not susceptible to degradation from FAC.

At the end of 1996, industry initiated efforts to develop a program to address erosion-corrosion. These initial efforts were directed at single-phase systems. Initial inspections were completed on all single-phase systems by 1989. Erosion-corrosion programs were in place on both single and two-phase by 1990 [11]. Since that time, service experience (Fig. 6) suggests that the number of failures due to erosion-corrosion has been reduced.

EPRI report NSAC/202L [12] provides general guidelines for the identification and inspection of components subject to FAC degradation.

Corrosion Attack in Service Water Systems. Uniform corrosion attack in service water piping, microbiologically induced corrosion (MIC), crevice corrosion, and pitting were typical causes of failure events of pipe components grouped in this category. Of these, MIC is the predominant corrosion mechanism in

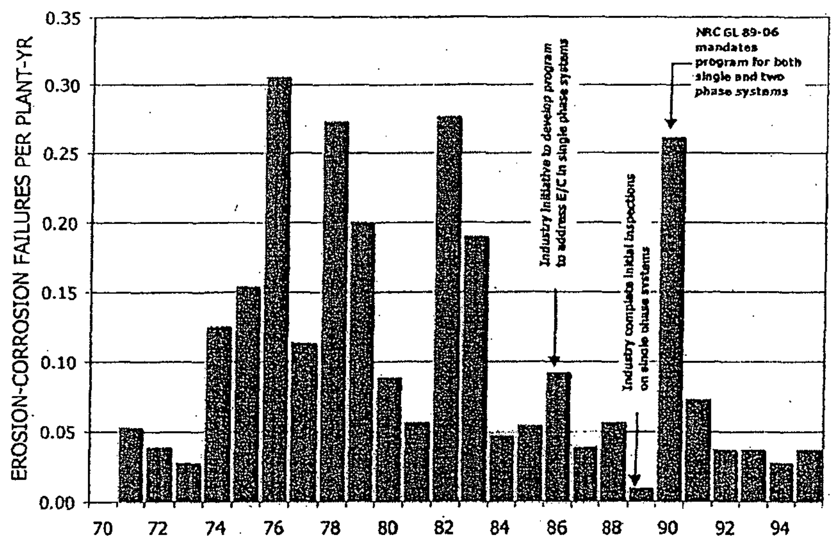

Fig. 6 Erosion-corrosion failures per plant year

these systems. In MIC, microbes, primarily bacteria, cause widespread damage to low-alloy and carbon steels. Similar damage has also been found at welds and heat-affected zones for austenitic stainless steels. Piping components with fluids containing organic material or with organic material deposits are most susceptible to MIC. The most vulnerable components are raw water systems, storage tanks, and transport systems. Systems with low to intermittent flow conditions, temperatures between 20–120°F and pH below 10, are primary candidates.

In response to NRC Generic Letter 89-13, industry was instructed to implement a comprehensive program to address corrosion in service water systems. Prior to this, the service water integrity programs relied on the Section XI periodic leak and hydrostatic pressure test requirements. Under the Section XI program, the service water system integrity management approach was "reactive" in nature; that is, corrective action was taken when damage was sufficient to result in visible leakage. The Generic Letter 89-13 augmented programs require plants to take a more "proactive" approach to the problem. For example, many programs implemented improved chemistry control to mitigate the establishment of MIC sites, volumetric inspections (UT/RT examinations), and component condition monitoring and trending.

EPRI reports TR-103403 [13], NP-5580 [14], and NP-6815 [15] provide additional information regarding MIC degradation.

High-Cycle Mechanical Vibration Fatigue. More and more attention has recently been paid by operating plants to prevent unexpected piping failures due to high-cycle vibration fatigue. Small-bore pipe (<1 in. NPS) socket-welded vent and drain connections in the immediate proximity of vibration sources tend to be most susceptible to this failure mechanism [16–18]. Unlike the previously discussed mechanisms, vibration fatigue does not lend itself to periodic inservice examinations (i.e., volumetric, surface, etc.) as a means of managing this degradation mechanism. The nature of this mechanism is such that, generally, almost the entire fatigue life of the component is expended during the initiation phase. Once a crack initiates, failure quickly follows. Therefore, the absence of any detectable crack may not assure reliable component performance. In addition, for many of these components, the plant conditions when vibration levels are unacceptable may be very difficult to predict and limited to short time periods of unique plant/system configurations. This would explain why we continue to observe cases where vibration fatigue failures occur late in the plant's operating life [8]. Therefore, the fact that a vibration failure has not occurred within the first few years of plant operation may not preclude future failures.

Figure 7 shows the number of pipe failure events per reactor plant-year reported to NRC as being caused by high-cycle vibration fatigue. Prior to 1976 piping vibration fatigue was addressed

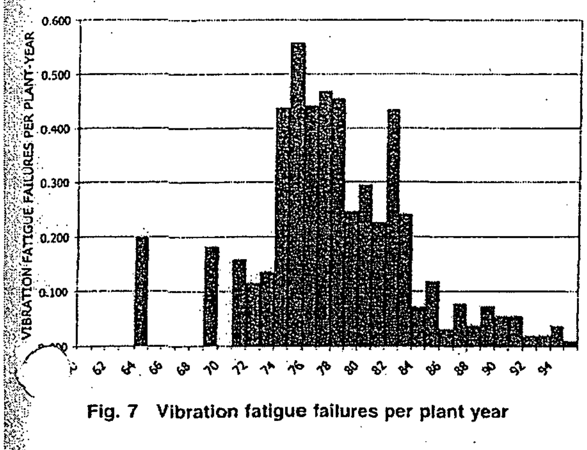

Fig. 7 Vibration fatigue failures per plant year

as a result of a problem resolution. Between 1976 and 1982, the significant amount of vibration fatigue related failures (Fig. 7) fostered increased attention to this problem by code and regulatory bodies. The NRC incorporated requirements to perform vibration testing as part of nuclear power plant initial testing programs [16], and by 1982 the ASME published an operating and maintenance standard [19] which specified requirements for preoperational and initial start-up vibration testing in nuclear power plants.

Risk-Based Inservice Inspection. Service experience and the augmented inspection programs have demonstrated a need on Section XI's part to move in new directions and shift its emphasis away from simple inservice "inspection" rules to establishing effective integrity management programs for nuclear plants. Ideally, these new programs should include the following characteristics:

1 Future programs need to be based on an understanding of failure mechanisms and focus attention on the locations in the plant system most likely to be affected by these mechanisms. This will allow plants to identify problems in a proactive manner, so that corrective actions can be planned and implemented before failures occur.

2 Monitoring and inspection methods need to be designed specifically for the degradation mechanism of concern. This has been referred to as "inspection-for-cause."

3 The integrity management program should be designed to ensure reliable component operation. For example, inspection frequencies may need to be adjusted to ensure that the failure probability of the component is maintained at an acceptable level. ASME Section XI hopes to accomplish these objectives moving in the direction of risk-informed inservice inspection (RISI).

As a first step, ASME Section XI has recently developed pilot code cases that allow for the use of alternative RISI rules for piping. These code cases grew out of work sponsored by ASME research [20] and EPRI [21]. The three code cases implementing this technology have been incorporated into ASME Section XI Code Cases N-560, N-577, and N-578. These initial efforts focused primarily on the identification of inspection locations and the implementation of appropriate inspection methods. Industry pilot applications [22,23] have been completed for each code case. Each application has been reviewed and approved by the NRC for consistency with NRC guidelines [24].

Probabilistic-Based Inspection Strategies

Thus far success of the initial RISI studies has been measured in terms of estimated reductions in nuclear power industry and regulatory burden, anticipated man-rem exposure reductions and calculated improvements in reactor safety. These improvements in safety have assumed that the selected inspection locations are examined using reliable NDE methods at appropriate frequencies in order to achieve a reduction in failure probabilities. In the long run, ultimate success will be seen in a reduction in the occurrence of piping leaks in these systems. Therefore, future inspection strategies will need to manage component failure frequencies.

In this section we show how a probabilistic approach can be applied to determine inspection frequencies that account for demonstrated NDE performance and ensure reliable piping performance is maintained throughout the component's original or extended operating life. In the example described in the forthcoming, we assume that the weld location is subject to thermal fatigue. The inspection frequency necessary to maintain the component's failure probability at or below that associated with the fatigue limit specified in the original construction ASME Section III design code (e.g., cumulative usage factor (CUF) must be less than unity) is then determined.

Probabilistic Approach. Probabilistic fracture mechanics calculations are presented to demonstrate that an augmented level

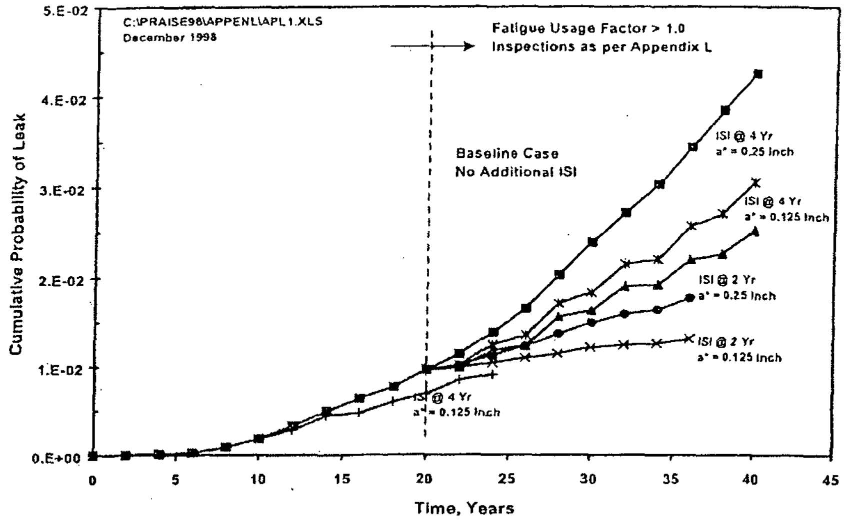

Fig. 8 Calculated probability of leak before and after implementation of inspection program

of inservice inspection can ensure that failure rates of fatigue critical components should not increase as operation is continued beyond usage factors permitted by the design code. Uncertainties in flaw growth rates and in flaw detection were addressed by application of the probabilistic fracture mechanics code pc-PRAISE [25]. Suitable inspection frequencies were established for a given flaw detection capability (probability of detection or POD curve) by adopting a goal for an acceptable piping failure probability (i.e., probability of through-wall crack per weld per year). Continued operation for calculated CUFs exceeding unity was taken to be acceptable only if additional inspections are performed. These inspections are required to maintain calculated failure rates at levels less than (or equal to) calculated failure rates before the usage factors became unity.

Probabilistic Calculations. The example considers a stainless steel pipe (29-in. outside diameter by 2.5-in. wall) which is loaded at 5000 cycles per year to give a CUF=1.0 after 20 yr of operation given a weld root stress concentration factor of 3.0. This corresponds to a nominal alternating stress of 27.3 ksi and a peak alternating stress, at the weld root, of 81.9 ksi.

The pc-PRAISE model assumed semi-elliptical surface flaws with aspect ratios of 12 and 20, and a Paris law for fatigue crack growth having a mean rate corresponding to constants of $C = 9.14\mathrm{E}{-}12$ and $m=4$. A simplified treatment of flaw initiation was assumed. At time=0.0, very small inner surface cracks were assumed to be present, with depths uniformly distributed between 0.005–0.010 in.

The alternative inspection frequencies were limited to the case of no inspections and inspections every 2 or 4 yr, with the inspection program being introduced after 20 yr of operation. The reliability for the ultrasonic NDE was described by the error function-type curves used by the pc-PRAISE code to describe flaw detection. Two bounding curves were assumed for purposes of the demonstration calculations. The less effective NDE assumed a threshold detection capability (50 percent POD) for a 0.10-t flaw ($a^*=0.25$ in.), whereas the more effective NDE had a 50-percent POD for a 0.05-t flaw (0.125-in.). In each case, the POD curve provided significantly better detection capabilities for flaws of greater depths, such that flaw depths 0.25 and 0.50 in., respectively, or about twice the threshold size, could be detected with a probability of better than 90 percent.

Figure 8 shows the predicted cumulative probability of leak (through-wall crack) as a function of the operating time (0 to 40 yr). At 20 yr (when the calculated CUF becomes 1.0), the cumulative leak probability is about 1.0E-02, or one chance in 100 that the weld would fail. If no inspections are performed, the cumulative failure probability curve continues to rise and with an increasing failure rate. All of the alternative inspection scenarios (combinations of POD and inspection frequency) reduce the calculated failure probabilities, but some scenarios reduce the failure probability much more than others. The most effective inspection ($a^*=0.125$ in.) reduces the failure rate by about an order of magnitude compared to the alternative of no inspection. In this case the failure rates during the second 20 yr of operation are actually substantially lower than the corresponding rates during the first 20 yr of operation. Some of the other less rigorous inspections of Fig. 8 are also sufficiently effective to maintain the calculated failure rates at or below the rate that exists at the time (20 yr) when the CUF attains the limiting value of unity. For example, an Appendix L inspection with a 4-yr frequency and $a^*=0.125$ in. would meet the probabilistic criteria as well as the alternative of a 2-yr frequency with $a^*=0.25$ in. Therefore, in this extreme case where thermal fatigue loading is significantly high, a 2–4-yr inspection frequency will maintain the component's reliability at design basis levels.

Conclusions

The nuclear power industry has successfully implemented programs to manage degradation of pressure boundary components. These programs have focused on unexpected degradation mechanisms that have impacted plant operations well before the end of the expected plant design life. Programs have also been implemented to address potential mechanisms such as fatigue cracking that were identified as life limiting as part of the plant design basis. Monitoring of components in accordance with plant inservice inspections programs can ensure that the reliability of piping systems is maintained throughout the remaining design life, and address issues related to plant life extension beyond the original 40 yr of the original design.

Inspections at appropriate frequencies with reliable NDE methods can manage the potential degradation mechanisms, and thereby justify continued operation even when calculated design limits may be exceeded. It is even possible with an aggressive inspection program to decrease failure frequencies during the later periods of plant life to the same levels that existed relatively early in life.

By applying probabilistic methods, future inspection strategies cannot only be consistent with the service conditions and the demonstrated performance levels of the NDE methods, but will ensure that the reliability of the piping is maintained over periods of continued operation. Inspection strategies, designed in this fashion, will be a powerful addition to current risk-based ISI models.

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