Development of a Dynamic, Plant Condition-Dependent Probabilistic Safety Assessment

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Abstract: Although each nuclear power plant has a plant-specific probabilistic risk assessment (PRA) that reflects design differences from other plants, the condition of each plant changes uniquely with time. A great deal of surveillance data are collected for the plant that reflect the changing condition of the plant. In some instances, plant staff use these data to guide the plant's preventative maintenance and surveillance programs. In general, however, these data are not used to characterize the evolving risk of the plant. Our understanding of the underlying mechanisms for the degradation of systems, although far from perfect, is improving with time. The possibility of developing a condition-dependent PRA is explored that would take a first principles approach to modeling the progression of degradation mechanisms, periodically adapting the model to account for surveillance results, and using the model as a basis for a time-dependent characterization of plant specific risk. Because surveillance data would be used to periodically assess the consistency of the observed behavior with model predictions, it might be possible to provide early identification of unanticipated degradation mechanisms. A case study is described involving a potential bypass accident sequence involving the progression of flow-accelerated corrosion in secondary system piping and stress corrosion cracking of steam generator tubes.

Keywords: Probabilistic safety assessment, plant condition monitoring, dynamic risk assessment

1. INTRODUCTION

In recent years, with support from the nuclear industry, the U.S. Nuclear Regulatory Commission has fostered a proactive approach to the management of plant aging processes [1]. The objective of proactive materials degradation assessment (PMDA) is to focus surveillance and preventive maintenance on areas of the plant that are known to be susceptible to degradation processes. In many respects, while PRAs are intended to be plant-specific, they rely extensively on generic data bases. Although plant-specific data are often used to update generic data, a PRA is more representative of a population of similar designs than it is a plant-specific assessment based on that plant's observed conditions. Effectively, a PRA only goes skin deep rather than reflecting the observable measures of the condition of the plant such as the results of eddy current testing, ultrasonic testing, valve monitoring, and other monitoring techniques. A PRA that projects the condition-dependent risk of a plant could be an effective tool that would improve management of age-driven risk. In particular, it would provide strong support to PMDA. At this point, it should be indicated that while existing risk monitors somewhat reflect the current plant condition, the information that they display is resticted to only active components and does not include the condition of passive components.

Section 2 presents a process by which a dynamic assessment of the progression of degradation mechanisms allowing quantifying of time-dependent risk better reflects the evolving plant conditions by using the results of surveillance activities on passive components to continuously update the degradation models.The condition-dependent PRA of the plant could then be used as a tool for the management of surveillance and maintenance activities.

The case study described in Section 3 of this paper involves the mechanistic assessment of multiple degradation mechanisms. To determine their contribution to risk of the scenario, the frequency of the accident initiating event (a steam-line break), the failure on demand of an active system (main steam-

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line isolation valve), and the induced failure of degraded steam generator (SG) tubes due to the depressurization of the secondary side of the affected steam generator are all under consideration. The modes of degradation modeled are flow-accelerated corrosion (FAC) of the steam-line and primary water stress corrosion cracking (SCC) of the SG tubes (Alloy 600). The analysis is performed in a dynamic manner, in which the time-dependent evolution of degradation mechanisms is followed over the plant lifetime. The analysis includes assumed periodic surveillance activities with some probability of detection and repair depending on the extent of degradation and the reliability of the surveillance tools. If the secondary side of the steam generator depressurizes as the result of an unisolated steam line break, degraded steam generator tubes could fail. During each operating cycle, the probabilities are assessed of a spontaneous steam generator tube rupture (SGTR), a steam-line break resulting from FAC, the conditional probability of a MSIV failure to close, and the conditional probability of steam generator tube failure resulting from depressurization of the secondary side of the steam generator. The number of tubes plugged at the end of a cycle is also assessed.

Section 4 describes the approach and assumptions in a MATLAB model developed to project the time-dependent progression of component degradation and Section 5 presents the results of those analyses. Finally, in Section 6 conclusions are drawn regarding the feasibility and potential benefits of the concept of a living, condition-dependent probabilistic safety assessment.

2. DEVELOPMENT OF A CONDITION-DEPENDENT RISK MODEL

One of the limitations of PRA methodology as it has been applied since the mid-1970s is the static nature of the event tree/fault tree approach. Considerable research has been performed to explore the benefits of dynamic event trees (DETs) [2] to improve PRA predictions. Although some of the events analyzed in traditional static event trees naturally occur in an established order, that is not necessarily true for all cases. Particularly within the context of uncertainty analysis, the order of events can vary based on different sets of inputs [3]. DETs are not restricted by a priori assumptions about the order of events. They also enable analyses of accident progression to be performed in a manner that is mechanistically consistent with computer codes designed for transient analysis (e.g. RELAP [4], MELCOR [5]). DETs are an essential element of advanced human reliability analysis tools that attempt to simulate the cognitive response of human operators to the changing environment of an accident.

DET approaches have been examined for Level 1, 2 and 3 PRAs and dynamic tools, such as ADS [6], ADAPT [7], MCDET [8] have been developed to implement these approaches. In the Risk Informed Safety Margins Characterization Program [9], the U.S. Department of Energy (DOE) is extending the concept of dynamic analysis to plant performance assessment in a New Generation Systems Code (NGSC), also known as RELAP 7, capable of spanning the complete spectrum of time scales and multi-physics aspects of the time-dependent evolution of events in a nuclear power plant [9]. RELAP 7, for example, will not only address the plant's dynamic behavior within the time-scale of a loss of coolant accident, but should be able to assess slowly evolving changes in plant conditions during the fuel cycle leading up to the loss of coolant accident. It is within the conceptual framework of NGSC that we are considering a dynamic condition-dependent PRA. The other aspect of the dynamic condition-dependent PRA that is different from current practice is the very tight coupling between the evolution of the condition-dependent PRA and the co-evolution of surveillance and maintenance practices at the plant.

Clearly, there are practical limits today in the extent to which this type of dynamic condition-dependent PRA could be performed. The analyst would limit the scope of the effort to those mechanisms that are most likely to lead to failure, are sufficiently known to be able to make reliable predictions of degradation, and for which failures have risk significance. This paper describes a conceptual approach to converting an existing PRA to a dynamic, condition-dependent PRA that reflects component degradation. The approach in this study relies to a large extent on the results of expert elicitations documented in NUREG/CR-6923, "Expert Panel Report on Proactive Materials Degradation Assessment" by Brookhaven National Laboratory [1].

Degradation mechanisms affect the performance of both active and passive systems. In this paper, we focus on the impact of degradation of passive systems. In general, passive system failures can either be the initiator in the chain of events of an accident scenario or can affect the performance of the plant's systems, structures and components (SSCs) in responding to the event. Much of the focus of nuclear reactor regulation has been on the potential for pipe breaks in the reactor coolant system and the ability of safety systems to prevent core damage. The degradation mechanisms, particularly stress corrosion cracking of stainless steel, that can lead to the failure of primary system piping have been well studied [1]. The degradation mechanisms that affect the carbon steel portions of the power conversion system are different from those in the primary system. The rupture of secondary system piping not only could potentially lead to severe core damage but also represents a serious threat to plant operating personnel. Historically, FAC has been a major problem in the performance of components of the power conversion system including events leading to fatalities. As a result, all plants now have a FAC management program to address situations wherein vulnerable areas are identified and piping is replaced with steels that are less susceptible to FAC or surveillance is periodically undertaken.

Pipe breaks in either the primary or secondary system arising from corrosion mechanisms are found to be important initiating events in plant risk assessments. Degradation mechanisms could also affect the performance of safety systems that are intended to mitigate the severity of the scenarios that develop from the initiating event. For example, the containment structure of a plant provides the final barrier to the release of radioactive material to the environment in a severe accident. In a PRA, the failure probability and mode of failure of a containment are typically treated in a probabilistic manner as a function of internal pressure only. However, degradation of the containment shell can lead to failure at a lower pressure than for a pristine containment shell [10].

One measure of importance of a degradation mechanism is the extent to which it impacts core damage frequency (CDF). However, even if consideration were only given to degradation mechanisms that affect the frequency of initiating events, CDF does not fully characterize the risk significance of an event. For example, an initiating event leading to containment bypass is of much greater concern than a primary system LOCA within an intact containment. For this reason, a measure of the importance of a degradation mechanism is the potential impact on the consequences of a scenario in addition to its frequency. Very few full PRAs (Level 3 PRAs), which include the assessment of offsite consequence, have been performed for U.S. nuclear power plants. On the other hand, all nuclear power plants have limited Level 2 risk analyses that address the probability of different modes of containment failure, in particular the probability of a large early release of radioactive material to the environment. The results of the existing plant PRAs are used in risk informed regulatory activities through the risk meassures of CDF and of large early release frequency (LERF), which are considered surrogates for the NRC's probabilistic safety goals [12].

The modes and rates of degradation processes typically depend on the time-dependent thermal-hydraulic and stress environment to which the SSC is exposed. Figure 1 illustrates a process by which the condition-dependent behavior of the plant risk would be assessed. The time-dependent environment would be calculated using the *Plant Simulator*, for example the RELAP 7 code (or a legacy code such as RELAP 5 or MELCOR). At time t=0, plant condition, plant configuration and state of process variables (Initial Conditions) are fed into the *Plant Simulator*. Plant configuration and component failure rates/probabilities are also fed into the *PRA Code* for the prediction of *Risk Metrics and Importances* at $t=kT$. The variable T is a user specified time interval, possibly chosen to represent the duration of an operating cycle or a surveillance interval as well as to model degradation dynamics adequately and k is the number of the time interval. The *Plant Simulator* produces the thermal-hydraulic/neutronic/stress data which are fed into the *Component Aging Progression Model* which is assumed to stay constant within the time interval T to predict failure rates/probabilities at $t=kT$. The *Plant Simulator* in Figure 1 may be required to operate over distinctly different time scales such as: 1)

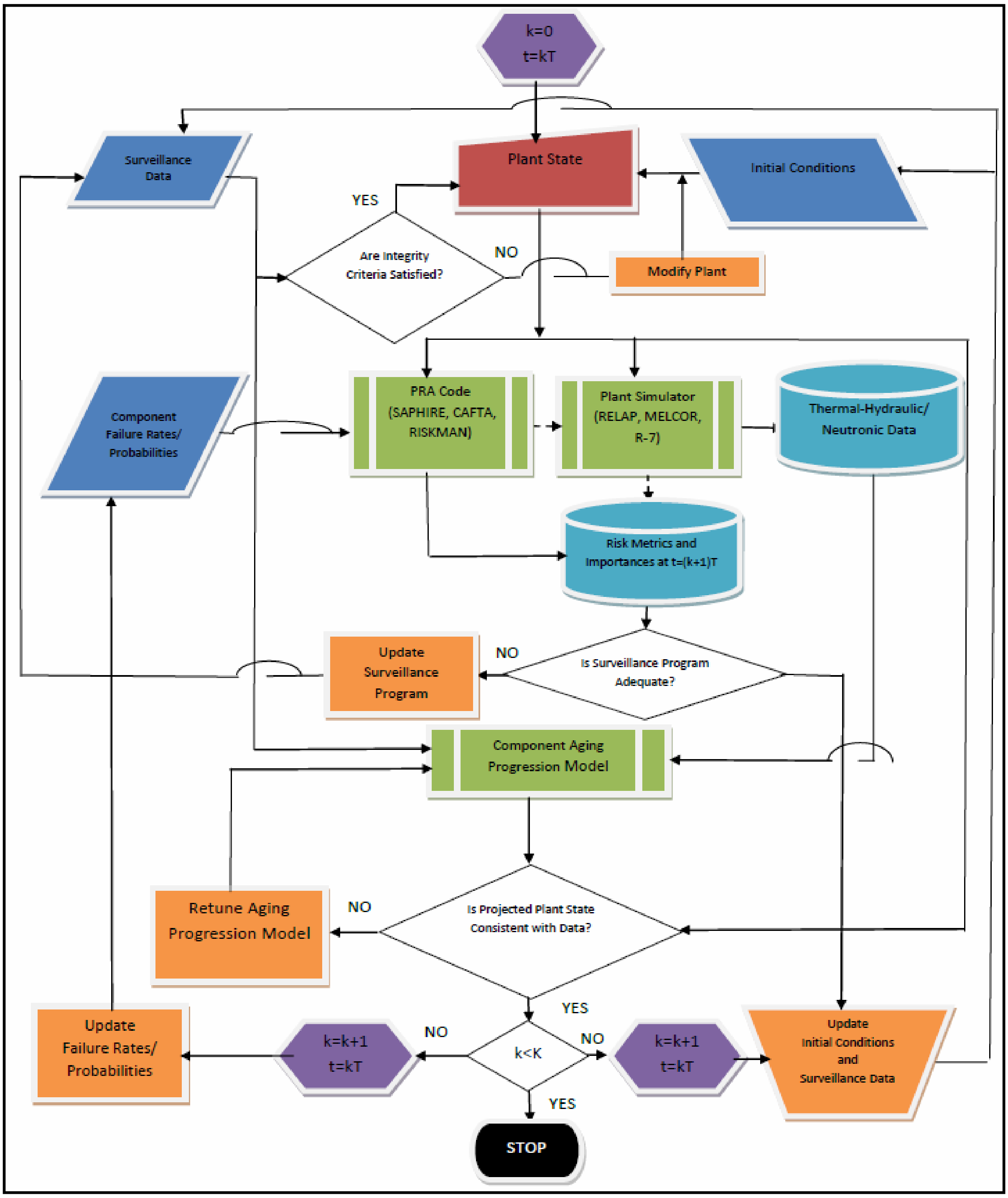

Figure 1: Evolution of Dynamic, Condition-Dependent PRA

the quasi-steady state condition while the plant is at power during a fuel cycle, and, 2) the dynamic time frame of a reactor shutdown and startup or the transient response of the plant to an accident. For the quasi-steady state condition, it is likely that the thermal-hydraulic conditions will be maintained constant based on the results of offline steady-state calculations performed with the *Plant Simulator*. *Initial conditions/surveillance data*, maintenance and repair actions affecting the plant state and *Component Failure Rates/Probabilities* inferred from these data are also updated at each time *kT*.

At the end of each time interval *T*, the plant condition is re-evaluated as it impacts the determination of the plant risk for that time interval. If the predicted *Risk Metrics and Importances* are found inadequate, the surveillance program is updated. As degradation processes continue over the time period, the potential of some kind of an initiating event such as a leak or rupture of a pipe at a weld will grow. Based on the condition, the likelihood of an initiating event of this nature will be determined, which will affect the plant risk for that time interval. Similarly, degradation will occur in components that need to operate in response to an initiating event, again affecting the outcome of the risk assessment. Thus, it is necessary not only to project degradation as a function of time but also to interpret the impact of a level of degradation on the probability of the occurrence of an initiating event or the impact of a level of degradation on the performance of a component. If the predicted *Plant Condition* is found to be inconsistent with surveillance data, then the aging progression model would be retuned. Similarly, the results of surveillance performed within a particular time interval could indicate the need to repair or replace a component or structure. Thus, components or structures can be returned to some initial state at which degradation mechanisms will again continue to degrade their performance. The time is incremented by T and the process is repeated until the target time horizon kT is reached.

For the update of *Initial conditions*/ *surveillance data*, the proposed approach would employ a model similar to that employed by [13] in that repair rate is

$$\omega = \frac{P_I P_D}{T_F + T_R} \tag{1}$$

where T_F is the mean time between inspections for flaws (i.e., the inspection interval), T_R is the mean time to repair once the flaw is detected, P_I represents the probability of inclusion or the probability that a piping element with a flaw will be inspected per inspection interval given that a certain percentage of elements are inspected. This concept is explored in detail by [14] and [15]. P_D is the probability of detection. The concept of modeling probability of detection for a specific testing method (e.g., UT) and as a function of the dominant physical parameter (e.g., crack length) has been developed in the literature [16-18]. Human factors are implicitly considered; however, some detection models provide for more explicit consideration. With the probability of detection often developed by testing performed on artificial defects, a Bayesian approach may be employed [19,20] to update "generic" data to reflect plant-specific surveillance testing techniques and results.

3. CASE STUDY

3.1. Scenario Description

The accident scenario selected for analysis involves failures of passive components due to degradation mechanisms for which there has been extensive operational experience. More specifically, in the case studied here, a steam line break outside the containment and downstream of a main steam isolation valve (MSIV) serves as the initiating event. Following the steam line break, the accident continues with a failure of a MSIV and the rupture of a flawed SG tube in the steam generator upstream of the MSIV. FAC of piping in the power conversion system and SCC of steam generator tubes were selected as the acting degradation mechanisms.

Steam lines penetrate the containment wallmaking it necessary for each line to contain an isolation valve external to the containment. The containment building becomes pressurized if the steam line breaks inside the building. This is a design basis accident analyzed in safety analysis reports (SARs). The containment integrity is not lost if the MSIVs close properly. If the depressurization of the SG leads to rupture of a tube, the reactor coolant system (RCS) begins to depressurize and it becomes necessary to inject emergency core cooling water into the RCS to prevent core damage. The conditional probability of core meltdown should be similar to that seen in small break LOCA scenarios. However, if the emergency core cooling system (ECCS) works in both injection mode and recirculation mode, core meltdown would be avoided.

If the steam line breaks downstream of the isolation valve, the situation is different. If the MSIV fails to actuate and a tube rupture is induced by the increased pressure differential, there is the potential for core meltdown with the containment bypassed. In this case, the ECCS is effective in the injection phase but fails due to lack of water in the sump when the switch to recirculation occurs. Although the likelihood of this scenario is expected to be very small, it is of interest because of the potential for containment bypass and a large late release of radioactive material to the environment. This type of event was at one time considered a generic safety issue (GSI-188) [21]. A variation of the scenario has been studied by Argonne National Laboratory in support of the resolution of that generic safety issue in which the failure mechanism of the steam generator tubes is associated with stresses in the tubes induced by differential expansion of structures in a steam generator and binding of tubes at tube support plates.

3.2 Degradation Mechanisms

3.2. 1. Steam Generator Tubes

The integrity of the SG is essential since a leak of a sufficient size or a rupture may contaminate the steam in the secondary loop and cause a release of radioactive material to the environment. To meet this challenge, special materials and heat treatments are used for the fabrication of SGs [22]. Many of the primary circuit components such as reactor pressure vessel (RPV), pressurizer, or the steam generator are produced from nickel-based alloys (referred by their trademark name as Inconels). These alloys have coefficients of thermal expansion similar to low alloy steels and are characterized by relatively high corrosion resistance in PWR primary and secondary water environments. The use of nickel-based alloys has evolved over the decades as new degradation mechanisms were identified.

In the early-to-mid 1970s, essentially all nuclear power plants (NPPs) in the United States used SGs with tubes made from mill-annealed Alloy 600 (600MA). However, the water chemistry was found to lead to excessive thinning. In response, NPPs changed their water control programs to eliminate this mechanism. In the mid-to-late 1970s, a process called denting resulting from the corrosion of the carbon steel support plates and the buildup of corrosion products in the crevices between tubes and tube support plates became a major issue. Although measures such as changes in the chemistry of secondary loop were taken to limit this problem, other mechanisms continued to cause cracking in plants with Alloy 600MA tubes. The need for extensive plugging of Alloy 600MA SGs forced the industry to begin to replace them with SG tubes made from high-temperature treated Alloy 600 (600TT). The replacement process began in the early 1980s and up until now no significant degradation issues have been observed. Nevertheless, beginning in 1989, NPPs began using SGs made from thermally treated Alloy 690, which is believed to be even more corrosion resistant due to its nearly doubled chromium content. The switch to Alloy 690 was accompanied by changes in corresponding weld metals. Alloys 152 and 52 replaced previously used weld Alloys 182 and 82.

In this study, only SCC is modelled as a SG tube degradation mechanism, specifically the initiation and growth of axial cracks in Alloy 600 MA tubes. Crack growth rate is assumed to follow the form of the Scott model:

$$\frac{da}{dt} = 2.8 \cdot 10^{-12} \cdot (K - 9)^{1.16} \tag{2}$$

where

$\frac{da}{dt} =$ crack growth rate (m/s)

$K =$ crack tip stress intensity factor ($MPa\sqrt{m}$)

Although crack initiation time is typically modeled using a Weibull Distribution, for this study, the time for the initiation of cracks is based on a log normal distribution, as measured by Staehle [25], which indicated a constant logarithmic standard deviation of uncertainty over the range of measured data. A large number of cracks are initiated in the first few cycles of operation after which crack initiation tapers off slowly. By the end of the 40-year lifetime, a majority of the tubes have been subjected to crack initiation. In this study, crack growth was treated by dividing the steam generator tubes into 20 groups, with different inherent crack growth rates. The crack growth rate for each group was determined empirically based on data on crack lengths collected at the Ringhals plant [26] after 11 years of operation, as illustrated in Figure 2. Correction factors for the 20 groups of tubes were applied to the general form of Scott's crack growth rate formula [24] to duplicate the distribution observed at the plant. Figure 3 illustrates crack length as a function of time after crack initiation for examples from the 20 groups. In reality, it is not possible to distinguish between crack initiation timing and crack growth rate from the Ringhals data. The manner in which these processes were treated in the current study is recognized to be speculative. However, the predicted size distribution at 11 years and the timing at which PWRs have reached the economic limit on plugged SG tubes are in reasonable agreement with actual experience.

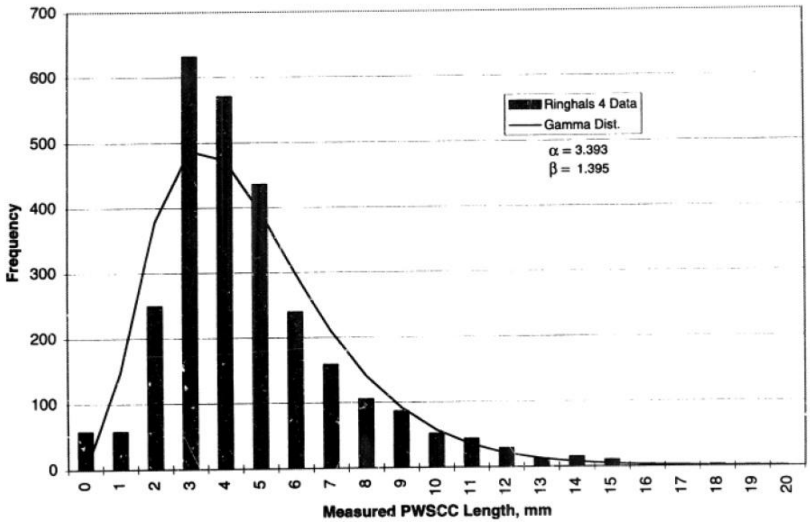

Figure 2. Measured Crack Length Distribution at Ringhals Plant [26]

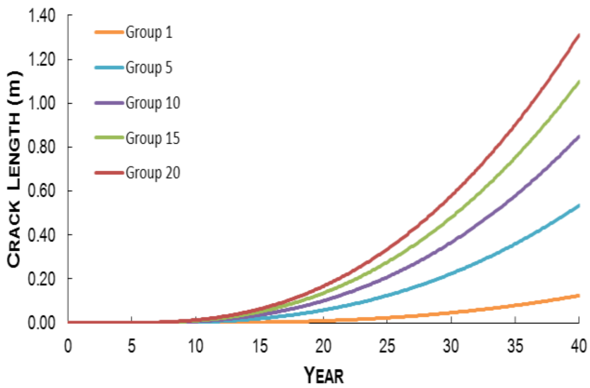

Figure 3. Crack Length as A Function of Time for Representative Groups

Data have been collected on the reliability of eddy current testing. Based on sets of repetitive tests by different teams of testers to determine the probability of failure to detect [27], it was assumed that the probability of failure to detect would be 0.02 at the first point at which a cohort of tubes reached the plugging criterion. Based on the binomial distribution, one can then determine the probability of different numbers of tubes that would be undetected from that cohort. To simplify the analysis, only the average number of undetected tubes were subsequently tracked in the dynamic analysis. At the next opportunity to detect, we have no data on which to assess whether an undiscovered tube that should have been plugged will now be detected. Thus, we used the simple approximation that at this point and subsequent tests, chances are that 50 percent of faulted tubes would be identified. Based on the binomial distribution, we can then determine the probability of one or more tubes remaining in a faulted state as a function of time:

$$P(x \geq 1|N,k) = 1 - (1 - p^k)^N \tag{3}$$

where
N = number of undetected tubes during the initial surveillance test
p = non-detection probability
x = number of undetected tubes
k = cycle of interest

3.2.2. Steam Line

Carbon steel piping of the secondary side carrying flowing water or wet steam is susceptible to FAC. This degradation process leads to wall thinning (material loss) that may result in pipe leaks or rupture. One of the most notable accidents caused by FAC occurred on December 9, 1986 at the Surry Nuclear Power Station. A rupture of an elbow in the condensate system resulted in four fatalities [28]. In PWRs, FAC has been only recorded in piping outside the containment [29]. Under normal conditions, a thin, protective layer of magnetite (Fe_3O_4) forms on the inside surface of carbon steel feedwater piping. Metal oxidation occurs at the metal-oxide interface in deoxygenated water. Ferrous species (Fe^{2+}) diffuse through the porous oxide layer and are dissolved into the water flow [28, 30]. FAC occurs in single- and two- phase flows. However, presence of water in its liquid state is necessary, and therefore it does not affect pipelines transporting dry or superheated steam [28].

The treatment of the probability of FAC leading to rupture in this paper is superficial relative to the effort that would be required in an actual risk assessment. The first step would be to examine plant drawings to determine locations on the secondary side where the conditions exist that could lead to FAC and which could not be isolated in the event of MSIV failure. The analyst would then examine the plant's FAC program to determine surveillance requirements for these locations. We also did not have data on the likelihood that an ultrasonic test of a vulnerable section of piping would not detect significant erosion of pipe wall thickness. Plant drawings were not available to us. Thus, we modeled a single characteristic location.

Industry and laboratory experience has led to the development of several semi-empirical models for predicting rates of FAC and resulting potential for pipe ruptures. The three most commonly used FAC models are contained within the WATHEC, CHECKWORKS, and BRT-Cicero packages [30]. The use of the latter two, however, is somewhat limited due to their proprietary nature. The KWU-KR model developed by Kastner and Riedle as part of the WATHEC code produced by Siemens/KWU is openly available and has been widely analyzed in published literature. For this reason, it was used for the simplified analysis performed for this paper. The KWU-KR model determines the rate of wall thinning due to flow-accelerated corrosion. This approach accounts for geometry, flow velocity, chemistry (pH and oxygen content), temperature, piping metallurgy (including chromium and molybdenum content of steel), and exposure time [30].

Failure of a component damaged by FAC can be calculated using a load-capacity formulation. In the case of feedwater carbon steel piping, the load is defined as pressure imposed on the piping during steady-state and transient conditions. The capacity is defined as the maximum pressure sustainable by a pipe subjected to wall thinning. A pipe fails when the load exceeds the capacity. NUREG/CR-5632 uses capacity expression from Wesley et al. shown in Eq. **Error! Reference source not found.** [29].

$$P_{capacity}(t) = \frac{\sigma_f \cdot h_{pipe}(t)}{[r + h_c(t)](1 + 0.25\varepsilon_f)} \tag{4}$$

where
$P_{capacity}(t) = \text{pressure capcity at time t (ksi)}$
$\sigma_f = \text{failure stress (ksi)}$
$r = \text{nominal pipe radius (cm)}$

$h_{pipe}(t) =$ pipe wall thickness at time t (cm)
$h_C(t) =$ calculated thickness of pipe corroded away at time t (cm)
$\varepsilon_f =$ median hoop strain at failure

For the purposes of this study, analysis was performed for two secondary-side pipe ruptures that have actually occurred. The results obtained with the KWU correlation were consistent with the actual plant experience. In these cases, failure occurred after approximately ten years of operation. Because we did not have the information required to examine multiple secondary side locations, for this analysis we assumed a ten year delay in operating history before a steam line break could occur but used the NUREG-1150 value for steam line break frequency after that period.

3.2.3. Main Steam Isolation Valve

Considerable data on the effects of aging and service wear on MSIVs, which have been collected in the National Plant Reliability Data System, are presented in NUREG/CR-6246 [31]. One of the purposes of a MSIV is to limit the consequences of a steam line break. In PWRs, MSIVs are typically located within a valve vault external to the containment boundary. Depending on the design of the MSIV, it may only stop flow in the downstream direction, providing protection for the turbine. For those designs, it is necessary to also include a check valve in the line that limits backflow for a break upstream of the valve. MSIV failure modes can be divided into six categories: failure to open, failure to close, spurious valve closure, spurious valve open, valve stem or shaft leakage, and valve seat leakage. The probability distribution of these different failure modes depends on the type of valve: gate valve, globe valve or check valve. Failure to close of a MSIV is the mode of concern analyzed in the accident scenario. Failure of the valve can either be the result of a fault in the valve or the valve actuator. NUREG/CR-6246 [31] presents failure data for the number of times that failure to close has been experienced for MSIVs. We did not have data on the associated number of challenges. Based on an estimate of three challenges per MSIV per year of plant operation in this period, we obtained a probability of failure on demand of 1.9×10^{-2} per demand, which is an unexpectedly high value. The value used in NUREG-1150 was 1×10^{-4} per demand.

3.3 PRA Model

The chosen scenario of concern is modeled using the Zion Nuclear Power Station as a reference PWR plant. The Zion plant was a two-unit station located on the shore of Lake Michigan. Each of the two units was a four-loop Westinghouse nuclear steam supply system with a rating of 1100 MWe housed in a large, pre-stressed concrete, steel-lined dry containment [32]. The safety functions and associated probabilities of failure on demand described in NUREG/CR-4550 Vol. 7 [33] were used to develop an event tree for the primary accident of interest resulting from a steam line break. The Top Events and corresponding failure probabilities are summarized in Table 1.

Table 1. Top Events and Failure Probabilities [33]

Top Event Acronym	Top Event	Probability of failure on Demand
SLB	Steam line break outside the containment (Initiating Event)	See Section 3.2
K	Reactor trip	1.8E-4
M	MSIV failure	See Section 3.3
TR	Tube rupture given pressure gradient	See Section 3.1
RW	Refueling water storage tank	2.4E-8
SS	Safety injection system actuation signal	2.2E-5
L1	Auxiliary feedwater actuation and secondary cooling	3.4E-5
HP	High Head Injection/Feed and Bleed	2.1E-8
R2	Low pressure recirculation	4.6E-4

In addition, the event tree was developed based on several assumptions:

- NUREG/CR-4550 Vol. 7 considers scenarios with combinations of AC buses not working. The scenarios considered here assume that all AC buses work properly due to the very low frequency of the scenario, in which any of the three AC buses is not working.
- MSIV failure leads to a blowdown of affected SG (secondary side) and depressurization of the steam header. If the faulted SG is one of the two lines that provide steam to the auxiliary feedwater system, only one auxiliary feedwater pump is available. In this case, the probability of failure of auxiliary feedwater increases.
- SGTR results in a small break LOCA.
- An operator can act to depressurize the RCS to slow down the release to the environment caused by leakage of RCS inventory into the turbine building. However, if the auxiliary feedwater system fails, it is necessary to rely on the feed and bleed system at high pressure.
- Since modern seals are less likely to fail, the scenario ignores the possibility of a pump seal LOCA.

The failure of reactor trip events (Anticipated Transients without Scram) were not analyzed because of their low probability. Of the remaining end states five result in severe accident conditions:

- SLB/MSIV/SGTR/failure of low pressure recirculation (late meltdown)
- SLB/MSIV/SGTR/failure of high pressure injection and feed and bleed (early meltdown)
- SLB/MSIV/SGTR/failure of auxiliary feedwater actuation and secondary cooling (early meltdown)
- SLB/MSIV/SGTR/failure of reactor safety actuation signal
- SLB/MSIV/SGTR/failure of refueling water tank flowpath (early meltdown)

Following a successful plant shutdown, the closure of a MSIV leads to a design basis accident resulting in a successful prevention of core meltdown. In case of MSIV failure, if the integrity of SG tubes remains uncompromised, a meltdown is also avoided. However, if the SG tubes rupture, a series of safety functions act to stop core meltdown. Failure of the first line of defense (i.e. supply of water from the refueling water storage tank or high pressure injection) leads to an early core meltdown. The last line of defense is the low pressure recirculation. In case of a SLB, there is no water in the sump that would permit recirculation to work properly. In that case, even if the system functions properly from a mechanical standpoint, it does not have the necessary inventory, which will result in a late core meltdown. The two remaining emergency actions that could be undertaken would include providing additional borated water to the refueling water storage tank or the containment sump.

4. MATLAB MODEL

4.1 Modeling Assumptions

A MATLAB model was written to follow the initiation and growth of cracks in steam generator tubes. It was assumed that the cracks were initiated in each cycle according to a log normally distributed initiation time as reported by Staehle [25]. This led to the initiation of cracks in 31% of the tubes by the end of the first two-year interval. The fraction of new tubes with newly initiated cracks then decreased rapidly with subsequent cycles. The crack growth rate of tubes was assumed to be independent of initiation time. Newly introduced cracks were assumed to have a length of 0.1 mm. Furthemore, their growth was characterized by a distribution of twenty growth rates which were followed in time. Cracks were assumed to have the length-to-depth ratio of 3:1 throughout their growth. It was assumed that when a crack had grown to 40% of the tube wall thickness, it would be plugged following the next inspection unless there was a detection failure. As discussed in Section 3.1, at the first time at which a cohort of tubes (a given initiation cycle and given growth rate equation) exceeded the plugging criterion, it was assumed that 2% of the tubes in that cohort were not identified as having exceeded the criterion. In subsequent inspections, the fraction of tubes exceeding the plugging criterion that should have been detected previously was reduced by 50%.

There are three critical crack lengths of importance: the length at which the 40% through-wall criterion is reached (1.5 mm), the length at which a steam line break with failure of an MSIV could potentially induce a rupture (33 mm) and the length at which the tube could rupture spontaneously (62.5 mm). Based on some analyses of characteristic pipes that had actually failed by FAC, it is assumed that a steam line break would not occur prior to the 5th two-year cycle. At that point the steam line failure frequency is taken as 1.88×10^{-3} per yr in agreement with the NUREG-1150 study.

4.2 Results

As expected, the model predicted rapid deterioration of SG tubes. By the 4th cycle (6 to 8 years), the cohort associated with cracks initiated in the first cycle and the most rapid growth rate had grown to the point at which plugging was required. Figure 4 shows the history of plugging for each of the four steam generators. There are 3,592 tubes per SG. Thus, if the limit of plugged tubes is 20% without substantially impacting plant performance, this limit would be reached in 10 years (5 two-year cycles). Historically, SGs with Alloy 600MA have lasted longer than 10 years but substantially less than the 40-year lifetime of the plant, indicating that the degradation model somewhat underestimating the time to initiate cracks or over-estimating the crack growth rate.

Because the model used to assess the probability of failure to detect cracks larger than the plugging criterion always has a non-zero probability at some point, SG tube ruptures would also occur with some probability. Figure 4 indicates the probability of SG tube rupture per SG as a function of time. The model shows that the SGs would probably be replaced before being susceptible to a high likelihood of tube rupture. Although ruptures have occurred in steam generator tubes in U.S. PWRs, they have not necessarily ruptured as a result of SCC.

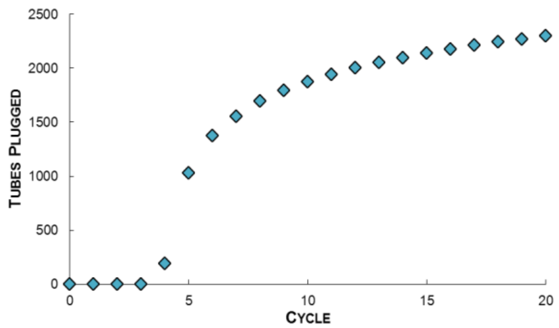

Figure 3. Number of Plugged Tubes Per SG as a Function of Time

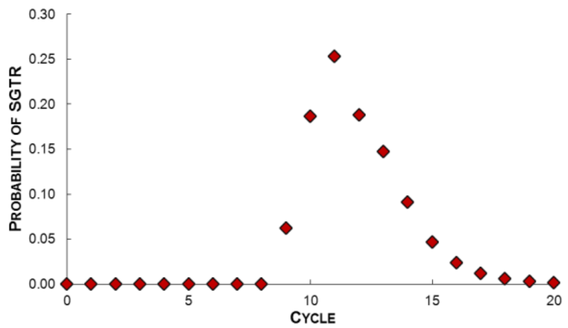

Figure 4. Probability of SG Tube Rupture vs. Cycle

Figure 5 illustrates the bottom-line objective of the analysis, the time-dependent risk of a SLB/MSIV/SGTR event. Because the critical length at which a tube becomes susceptible to failure resulting from depressurization of the SG is shorter than the critical length for a spontaneous rupture, the time at which the scenario probability becomes non-zero occurs earlier than the time at which an SGTR becomes non-zero. The time-dependent core damage frequency for this scenario peaks at 20 years. The average risk over the lifetime of the plant is 2.5×10^{-5} per yr. As indicated for the spontaneous steam generator tube rupture frequency, based on the results of this degradation model, the plant would probably have changed SGs before these conditions were reached.

5. CONCLUSIONS

The key questions regarding whether a dynamic, condition-dependent approach to PRA should be pursued are:

1. Is it technically feasible based on the state of the art of modeling degradation phenomena?

2. Is the level of effort required cost-effective?
3.Is the level of effort warranted, i.e would we believe the results sufficiently to use them in managing aging risk. Could the results help to identify unsuspected mechanisms or vulnerabilities?

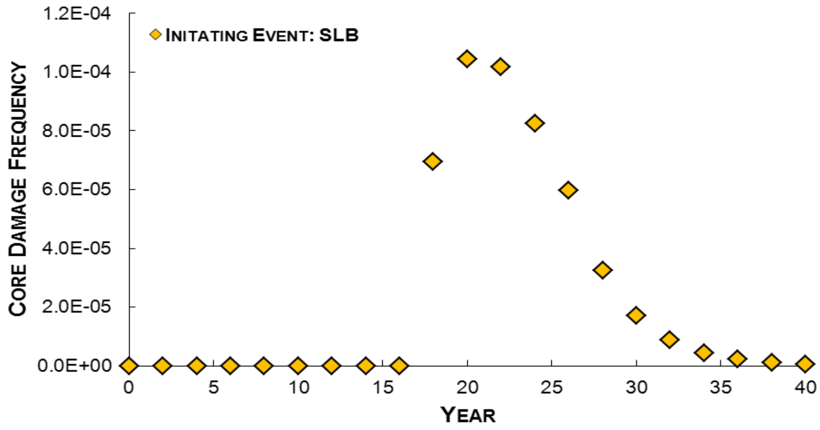

Figure 5. Core Damage Frequency for SLB/MSIV/SGTR Events

The results of the case study performed indicate that the mechanics of performing a condition-dependent risk assessment are feasible. The level of effort required would be substantially greater than for the case study because it would be necessary to identify a number of vulnerable locations in the plant and to consider alternative degradation mechansisms. There would also be a continuing effort associated with the analysis of surveillance results and modification of the parameters in the degradation model.

The case study involving Alloy 600 MA steam generator tubes was retrospective in nature. Thus, there were data available to initially tune the parameters in the degradation models and some indication of the dominant types of degradation mechansisms. In comparing the results of the degradation model for the case study with actual PWR operating experience in terms of the number of tubes plugged, the degradation model appears to have been slightly conservative. However, the model could certainly have been further tuned to be more consistent with operating experience. If we were to do a prospective analysis for the future risk associated with Alloy 690 SG tubes, we would not have a good starting model. The general form of the model might be the same as used for the case study, for example the Scott model. After each set of surveillances the model would be updated. Thus, in a boot-strap approach, by the time degradation mechanisms became important, the predictive capability of the model should be substantially improved.

An important consideration highlighted by the case study is the need for a quantitative understanding of the probability of failure of surveillance tools. This is an area in which more research is needed.

The performance of this very limited case study does not provide a definitive answer to the questions posed at the beginning of this section. Regardless whether the goals of developing a dynamic, condition-dependent risk capability for each plant are achievable today, the concept has significant potential value to risk managmeent and should continue to be pursued.

6. ACKNOWLEDGEMENTS

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