Implementation of Condition-Dependent Probabilistic Risk Assessment Using Surveillance Data on Passive Components

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Abstract

A great deal of surveillance data are collected for a nuclear power plant that reflect the changing condition of the plant as it ages. Although surveillance data are used to determine failure probabilities of active components for the plant's probabilistic risk assessment (PRA) and to indicate the need for maintenance activities, they are not used in a structured manner to characterize the evolving risk of the plant. The present study explores the feasibility of using a condition-dependent probabilistic risk assessment (PRA) framework that takes a first principles approach to modeling the progression of degradation mechanisms to characterize evolving risk, periodically adapting the model to account for surveillance results. A case study is described involving a potential containment bypass accident sequence due to the progression of flow-accelerated corrosion in secondary system piping and stress corrosion cracking of steam generator tubes. In this sequence, a steam line break accompanied by failure to close of a main steam isolation valve results in depressurization of the steam generator and induces the rupture of one or more faulted steam generator tubes. The case study indicates that a condition-dependent PRA framework might be capable of providing early identification of degradation mechanisms important to plant risk.

Keywords: Probabilistic risk/safety assessment, plant condition monitoring, dynamic risk assessment, condition-dependent PRA

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1. INTRODUCTION

In recent years, with support from the nuclear industry, the U.S. Nuclear Regulatory Commission (NRC) has fostered a proactive approach to the management of plant aging processes [1]. The objective of this 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 probabilistic risk assessments (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. A PRA effectively provides only a surface level treatment of plant's risk 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 time-dependent risk and provide strong support to PMDA.

One of the limitations of the traditional PRA methodology towards a condition-dependent is the static nature of the event tree/fault tree approach. Although some of the events analyzed on traditional static event trees naturally occur in an established order, that is not necessarily always the case. Particularly within the context of uncertainty analysis, the order of events can be different among different sets of inputs [2]. Considerable research has been performed to explore the benefits of dynamic event trees (DETs) [3] to overcome the limitations associated with the static nature of the traditional PRA. DETs are not restricted by a priori assumptions about the order of events. They also enable probabilistic assessment 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], MCDDET [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 in a New Generation Systems Code (NGSC), also known as RELAP 7, capable of spanning the complete spectrum of timescales and multi-physics aspects of the time-dependent evolution of events in a nuclear power plant [9]. The RELAP 7 code, for example, will not just address the plant's dynamic behavior within the timescale of a loss of coolant accident (LOCA), but should be able to assess slowly evolving changes in plant conditions during the fuel cycle leading up to the LOCA. It is within the conceptual framework of NGSC that we are considering a condition-dependent PRA that is dynamic in nature. The other aspect of the condition-dependent PRA that distinguishes it 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.

This paper expands upon the work presented in [10] to illustrate how an existing PRA can be converted to a condition-dependent PRA that evolves with time by utilizing the surveillance data on passive components. The approach 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] to identify risk significant components and degradation modes. In the present study, the DET simulation is associated with the prediction of the initiation and growth of cracks in steam generator (SG) tubes during the lifetime of the plant, with the

associated thermal-hydraulic conditions assumed to be constant during each operating cycle. However, for the more general case, the framework allows quantifying the impact of local thermal-hydraulic conditions, stresses and fatigue cycles on the initiation and propagation of flaws.

Section 2 of the paper describes the framework for the condition-dependent risk assessment model. The case study under consideration and its implementation are described in Section 3. Section 4 gives the conclusions of the study.

2. FRAMEWORK FOR A CONDITION-DEPENDENT PRA MODEL

Section 2.1 provides a more detailed background on the need for condition-dependent PRA. Section 2.2 describes the framework.

2.1. Background

Degradation mechanisms affect the performance of both active and passive systems. Active system degradation and replacement within the PRA context can be currently accounted for using risk monitors [11]. While traditional risk monitors provide only a point-in-time estimate of risk based on plant's configuration, prognostics algorithms that take time-dependent failure information of a given component to produce enhanced risk monitors have also been proposed (e.g. [12]).

In the context of plant life extension of traditional light water reactors (LWRs) from 60 to 80 years, degradations of plant's passive systems, structures, and components (SSCs) gain a high significance. Much of the focus of nuclear reactor regulation to date has been on the potential for pipe breaks in the reactor coolant system (RCS) and the ability of safety systems to prevent core damage. The degradation mechanisms, particularly stress corrosion cracking (SCC) of structural materials of the primary coolant system, that can lead to the failure of stainless steel primary system piping have been well studied [1]. The degradation mechanisms that affect the carbon steel portions of the power conversion system, such as secondary system piping, are different from those that affect primary system piping. Historically, flow-accelerated corrosion (FAC) has been a major problem in the performance of components of the power conversion system including events leading to fatalities of plant staff [13, 14]. As a result, all plants now have a FAC management program to address cases in which 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 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 materials 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. Degradation of the containment shell can lead to failure at a lower pressure than for a pristine containment shell [15].

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 would be 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. While 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, very few full PRAs (Level 3 PRAs), which include the assessment of offsite consequence, have been performed for U.S. nuclear power plants. In that respect, the results of the existing plant PRAs are used in risk-informed regulatory activities through the risk measures of CDF and of large early release frequency (LERF), which are considered surrogates for the NRC's probabilistic safety goals [16].

2.2. The Framework [10]

The modes and rates of degradation processes typically depend on the time-dependent thermal-hydraulic and stress environment to which the SSCs are exposed. Figure 1 illustrates the process by which the condition-dependent behavior of the plant risk could be assessed. The time-dependent environment would be characterized 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 and to model degradation dynamics adequately, whereas k is the number of the time interval. The *Component Aging Progression Model* incorporates first principles models for aging degradation of components. These models will typically include some tunable parameters. The *Plant Simulator* produces the thermal-hydraulic/neutronics/stress data, which are fed into the *Component Aging Progression Model*, which projects the extent of additional degradation expected over the next time interval T including the potential for component failure during that time interval. The combination of the *Plant Simulator* and *Component Aging Progression Model* provide the dynamic analysis of the plant condition. The *Plant Simulator* in Fig. 1 may be required to operate over distinctly different timescales such as: 1) 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 ($k=0, 1, \dots, K$).

At the end of each time interval T , the effect of aging processes is assessed through surveillance and the plant condition is reevaluated to determine the current and projected plant risk. Branches in Fig. 1 may lead to the initiation of passive component failure during the preceding cycle with an associated probability depending on the extent of component degradation. If the predicted *Risk Metrics and Importances* are found unacceptable, the surveillance program may be modified (*Update Surveillance Program*) or changes may be made to the plant (*Modify Plant*). 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 *Projected Plant State* is found to be inconsistent with surveillance data, then the aging progression model would be retuned. It is through this process that the approach

could identify the existence of a new degradation process or one in which conditions have resulted in accelerated degradation for some unknown reason. 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 repaired state (*Modify Plant*) at which degradation mechanisms could again be initiated. The time is incremented by T and the process is repeated until the target time horizon KT is reached.

In the process described in Fig. 1, the progression of aging is performed mechanistically, solving the equations describing aging mechanisms over each operating cycle (e.g. see Section 3.2). The results of surveillances performed during the outage are used to undertake repairs. The overall plant risk is then reassessed at the end of each interval. For those surveillances and repairs that occur during the course of a cycle, a model similar to that employed by [17] for repair branching can be used with a repair rate

$$\omega = \frac{P_I P_D}{T_F + T_R} \quad (1)$$

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 and P_D is the probability of detection. This concept is explored in detail in [18] and [19].

Methodologies to estimate probability of detection for a specific testing method (e.g. ultrasonic testing) and as a function of the dominant physical parameter (e.g. crack length) have been developed in the literature [20-22]. Human factors are implicitly considered; however, some detection models provide for more explicit consideration. On the other hand, with the probability of detection often developed by testing performed on artificial defects under controlled environments, it is not clear how the local conditions can affect the detection process, as well as propagation of errors. A Bayesian approach may be employed [23, 24] to update industry-wide data to reflect plant-specific surveillance testing techniques and results. This paper explores the path shown in red lines in Fig. 1 with $T=2$ years and $K=20$.

3. CASE STUDY

Sections 3.1 and 3.2 describe, respectively, the scenario and degradation mechanisms under consideration. The condition-dependent PRA model is examined in Section 3.3. The results are presented in Section 3.4. The scenario described in Section 3.1 was selected to illustrate the methodology of Fig. 1 because it involves an aging mechanism leading to a passive system failure as an accident initiator, an aging mechanism resulting in the vulnerability of a passive system given an initiating event and an independent component failure that could also result from aging-related degradation.

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 (SLB) 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 to close, depressurization of the secondary side of the steam generator upstream of the MSIV, and the rupture of one or more flawed SG tubes. FAC of piping in the steam line and SCC of SG tubes were selected as the acting degradation mechanisms.

Steam lines penetrate the containment wall, which makes 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 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 for this type of scenario should be similar to that calculated for 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 could be 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) [25]. A variation of the scenario was studied at Argonne National Laboratory in support of the resolution of that generic safety issue in which the failure mechanism of the SG tubes is associated with stresses in the tubes induced by differential expansion of structures in a SG and binding of tubes at tube support plates [26].

3.2. Degradation Mechanisms

3.2.1. SG 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 [27]. Many of the primary circuit components such as the SG tubes are produced from nickel-based alloys (referenced 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 pressurized water reactor (PWR) primary and secondary water environments [1]. The use of nickel-based Alloy 600 has evolved over the decades as degradation problems with this alloy were experienced.

In the early-to-mid 1970s, essentially all nuclear power plants in the United States used SGs with tubes made from mill-annealed Alloy 600 (600MA). However, the water chemistry was found to lead to heavy wastage. In response, the plants 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 degradations have been observed. Nevertheless, beginning in 1989, nuclear power plants began using SGs made from thermally treated Alloy 690, which is believed to be even more SCC 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 [27].

In this study, only primary water SCC (PWSCC) is modeled as a SG tube degradation mechanism. Although circumferential cracks can also be of concern, this study has been limited to the initiation and growth of axial cracks in Alloy 600MA tubes. Crack growth rate is assumed to follow the form of the Scott model [28] shown in Eq. (2).

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

where

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

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

a = crack dimension in the direction of crack growth

The challenge with using Eq. (2) is related to the crack tip stress intensity factor, K, which depends on crack size, applied stress and the geometric factor. Various fracture mechanics references provide formulas to calculate K based on semi-empirical and empirical models. SG tube flaws analyzed in this study were assumed to be semielliptical surface cracks (see Section 3.3.2), for which Newman and Raju developed an empirical stress intensity factor equation as $K = F\sigma\sqrt{\pi a}$, where F is geometric factor and σ is the stress at the crack tip [29]. Alternatively, a fracture analysis simulation may provide a better approximation of a real-life crack growth progression by accounting for various crack-containing plate geometries, time dependence of stresses at crack tip, and evolving crack geometry, where the ratio of crack depth-to-width is not constant with time.

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 [13]. In PWRs, FAC has been only recorded in piping outside the containment [14]. Under normal conditions, a thin, protective layer of magnetite (Fe_3O_4) forms on the inside surface of carbon steel feedwater piping. Ferrous species (Fe^{2+}) diffuse through the oxide layer and are dissolved into the water flow [13, 14]. FAC occurs in single (liquid) and two-phase flows. However, for FAC to occur presence of water in its liquid state is necessary, and therefore it does not affect pipelines transporting dry or superheated steam [13].

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 of success during ultrasonic testing of vulnerable sections of piping suffering from erosion of wall thickness. Plant drawings were not available to us. Thus, we modeled a generic location based on information available on secondary piping that had actually failed during operation.

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 [14]. 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 FAC. This approach considers pipe geometry, flow velocity, chemistry (pH and oxygen content), temperature, piping metallurgy (including chromium and molybdenum content of steel), and exposure time [14].

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. (4) [30].

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

where

$P_{\text{capacity}}(t)$ = pressure capacity at time t (ksi)

σ_f = failure stress (ksi)

r = nominal pipe radius (cm)

$h_{\text{pipe}}(t)$ = pipe wall thickness at time t (cm)

$h_c(t)$ = calculated thickness of pipe corroded away at time t (cm)

ε_f = median hoop strain at failure

In NUREG/CR-5632 analysis was performed on two secondary side pipe ruptures that occurred in reality [30]. The results obtained with the KWU correlation were consistent with the actual plant experience. In these two cases, failure occurred after approximately 10 years of operation. Because we did not have the information required to examine multiple secondary side locations, for this analysis we assumed a 10-year delay in operating history before a steam line break could occur, but used the NUREG-1150 [31] value of 1.88E-3 per year 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 by U.S. utilities, are presented in NUREG/CR-6246 [32]. 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 [32] presents failure data for the number of times that failure to close has been experienced for MSIVs. However, we did not have data on the associated number of challenges. The value used in NUREG-1150 is $1\text{E-}4$ per demand [31] and was adopted for this study.

3.3. Condition-dependent PRA Model

3.3.1. Reference Plant and Accident Progression

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 [31]. 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, as shown in Fig. 2. The Top Events and corresponding failure probabilities are summarized in Table 1.

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.
- SG tube rupture (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 the low conditional probability of $1.8\text{E-}4$ [34] for failure to scram. Of the remaining end states, five result in severe accident conditions:

- Steam line break (SLB)/main steam line isolation valve failure (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 flow path (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, which prevents proper recirculation. In that case, even if the system functions properly from a mechanical standpoint, it does not have the necessary inventory and will result in a late core meltdown. Two emergency actions that could be taken to prevent core meltdown include providing additional borated water to the refueling water storage tank or to the containment sump.

At this point it should be noted that while Fig. 2 depicts a traditional event tree, the branching data for the Top Event TR (0 or 1) depend on the state of SG tubes. Hence, the event tree in Fig. 2 changes structure as a function of the number plant cycles kT and the initiating event frequency becomes history dependent. In that respect, Fig. 2 represents only one element of a chain of event trees with changing structures which connect at End States #1 and #3 to the initiating event of the next cycle and which constitute the DET.

3.3.2. Crack Initiation, Propagation and Detection

In the case study, a 40-year plant lifetime is divided into $T=20$ two-year cycles. During each cycle, there exists a possibility that tubes may suffer from SCC manifested through development of cracks. Once a crack initiates, it may then grow and increase the susceptibility of tube rupture. This degradation may be, however, mitigated by a successful detection during a surveillance procedure. Tubes detected to have cracks may then be plugged. Although there are several approaches for improving the lifetime of SG such as water chemistry improvements or cleaning, tube plugging and sleeving are the most common methods used by plants [34].

The characterization of cracks in this study involves a number of approximations summarized in Table 2. First, the considered cracks were assumed semielliptical axial cracks with a constant depth-to-length ratio of 1 to 3 during their entire growth. A more accurate approach would have been to develop a distribution of aspect ratios based on data collected from SG tubes extracted from real plants. It is also assumed that each SG tube can contain only one dominant crack. In other words, cases of crack coalescence and multiple cracks in a single tube were not analyzed. The impact of these initial assumptions could be examined in a future study that would account for these simplifications with a more detailed statistical model. In this case, our study aims to examine the feasibility of using degradation models to create a condition-dependent risk assessment rather than to produce an actual risk assessment of a nuclear power plant.

In this study, crack evolution is considered in two separate stages: crack initiation and crack growth. Such a decision was made to overcome the lack of a single physical model explaining both phases of crack development. The first stage models crack initiation in which small cracks form on an inner surface of a SG tube subjected to a stress and corrosive environment. At the end of the initiation phase,

a crack is assumed to be 0.1 mm in length. Although crack initiation time is typically modeled using a Weibull distribution, in our study, this parameter is calculated from a lognormal distribution, based on the form of data presented by Staehle [35] and shown in Fig. 3, which indicates a constant logarithmic standard deviation of uncertainty over the range of measured data. Based on these data, a large number of cracks are initiated in the first few cycles of operation after which crack initiation tapers off slowly. More specifically, cracks are initiated in 31% of tubes by the end of the first two-year interval. By the end of the 40-year plant lifetime, 67% of the tubes have been subjected to crack initiation.

The second stage models crack propagation, which in this case was assumed to be governed by Eq. (2) and independent of initiation time. Based on the formula developed by Newman and Raju [29] presented in Section 3.2.1 and application of the parameters shown in Table 2, the geometric factor, F , needed to calculate the stress intensity factor (see Eq. (2)) should fall in a range from 0.72 to 1.04. In this study, a constant value of 0.93 was used without adjusting for the crack depth-to-wall thickness relationship. This was partly done to allow for slow initial crack growth rate and development of a crack length distribution observed at the Ringhals plant [36] after 11 years of operation, as shown in Fig. 4. Furthermore, crack growth was treated by dividing the SG 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 [36]. Normalization factors for the 20 groups of tubes were applied to Eq. (2) to duplicate the distribution of observed crack lengths at the plant based on the assumption that the lengths of cracks observed at Ringhals had strictly been the result of crack growth. This approximation is based on the assumption that crack growth rate at the plant was independent of crack initiation time. Fig. 5 gives the calculated crack length as a function of time after crack initiation for examples from the 20 groups. In reality, it is impossible to separate 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 approximate. However, we will show in Section 3.4 that the obtained results related to the timing of first plugging of tubes and time to SG replacement are in approximate agreement with U.S. operating experience.

The NRC requires that when the depth of a crack in an SG tube exceeds 40% of through-wall penetration the tube containing it must be plugged or sleeved [34]. Assuming a wall thickness of 1.27 mm [37], the plugging limit of 40% through-wall refers to a depth of 0.51 mm. Using a depth-to-length ratio of 1 to 3, this translates to a crack length of 1.52 mm. Based on information in Table 2 and Eq. (4), there are two additional critical crack lengths:

- the length at which a steam line break with failure of an MSIV could potentially induce a rupture (33.0 mm), and
- the length at which the tube would rupture spontaneously (62.5 mm).

Eddy current testing is performed on SG tubes during plant outages to determine the existence and size of flaws [38]. The sampling strategy used by the utility is established in the plant's technical specifications. Sample size and frequency of testing depend on the results of previous tests. Some 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 [39], it was determined that as the depth of the flaw approaches a 100% through-wall crack depth, the probability of failure to detect a flaw is approximately 2%. In this analysis, tubes are assigned to cohorts based on two criteria: a) the cycle in which the crack is initiated, and, b) the inherent rate of crack growth from the 20 crack growth rate categories. As each cohort reaches the point at which measurement would indicate that the tubes in that cohort should be plugged, a binomial distribution is used to determine a distribution of tubes from that cohort that fail to be detected and thus not plugged. A binomial distribution was

selected because the surveillance process is assumed to have an end state characterized by two possible outcomes (successful detection or failed detection). Since we have no data to assess whether an undiscovered tube in at time kT in need of plugging will be detected at time $(k+1)T$, the probability of identifying and not identifying a faulted tube are assumed to be equal or 50%. A better quantitative understanding of the in-situ probability of failure of surveillance tools and propagation of errors through multiple inspections will remove the need for this assumption which significantly impacts the results. Using the binomial distribution, we can then determine the probability of one or more tubes remaining in a faulted state as a function of cycle k as

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

where

N = number of undetected tubes during the initial surveillance test

p = non – detection probability ,

x = number of undetected tubes

k = cycle of interest

3.3.3. Summary of Assumptions

Since a number of assumptions were made in this study as indicated in the preceding sections either due to lack of data or to explore the path indicated in red lines on Fig. 1, these assumptions are summarized in this section for the reader's convenience:

- Only stress corrosion cracking is assumed to degrade the tubes.
- Only axial crack growth (down the length and across the thickness of the tube).
- Only single cracks are considered.
- A single aspect ratio of crack length to wall penetration (3:1) is considered.
- It is assumed that the distribution of crack lengths is solely the result of factors that affect the rate coefficient in the Scott formulation for crack growth rate.
- It is assumed that cracks are initiated with a single length according to an empirically determined crack initiating rate.
- It is assumed that there is no correlation between crack growth rate and crack initiation time.
- Following penetration of the tube wall, there is no assessment of leak rate or recognition that the total leak rate could exceed a technical specification limit, resulting in shutdown of the plant and the repair or plugging of tubes.
- Data on the reliability of eddy current testing was used to describe the testing conditions for crack depth characteristic of the plugging criterion.
- Analysis of flow-accelerated corrosion in secondary system piping is only used to establish the earliest time at which sufficient degradation of piping could occur to result in a steam line break.

Some of these assumptions could be refined with currently existing technology and better understanding of probability of failure of detection through additional plant surveillance data. Although some of the assumptions are significant, they do not affect the main goal of this paper (i.e. to illustrate how the surveillance data on passive components can be incorporated into the condition-dependent PRA), however, help to clarify the incorporation process.

3.4. Results

Using the approach described in Section 3.3, it was observed that by the fourth cycle ($k=4$ or 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 (i.e. crack depth has exceeded 40% of tube wall thickness). Fig. 6 shows the history of plugging for each of the four SGs. There are 3,592 tubes per SG. Thus, if SG efficiency is substantially deteriorated after 20% of tubes are plugged [40], then in our study this limit would be reached in 10 years (5 two-year cycles). Considering that: a) tubes installed in U.S. nuclear steam generators placed in service in the 1960s and 1970s were usually only mill-annealed Alloy 600, b) mill-annealed Alloy 600 SG tubes began exhibiting degradation in the early 1970s [41], and, c) first fully commercial PWR (250 MWe Yankee Rowe) was started up in 1960, this result is slightly conservative, but fairly consistent with operating experience. The underestimation of the time of tube failure (i.e. 10 years from the results of this study vs. 10-15 years from operational experience) could be due to the assumptions listed in Section 3.3.3.

Fig. 7 indicates the probability of at least one SG tube rupture per SG as a function of cycle following a SG depressurization induced by a SLB. As mentioned in Section 3.3.2., the critical length corresponding to a SG tube rupture susceptibility during a SLB is 33.0 mm. This probability is close to zero for the first seven cycles and is mainly a consequence of the time needed for the cracks to grow, high probability of detection, and the manner in which cracks are introduced. It takes some time, even for the tubes subjected to the most rapid crack growth rates, to reach rupture susceptibility. However, even before reaching this point, there is a likelihood of 98% that any given crack that has entered the propagation phase will be detected during the first inspection and then a 50% likelihood of detection during subsequent surveillance tests. Finally, as discussed earlier, most of the cracks are introduced to SG tubes during the early cycles as determined by the lognormal distribution. Therefore, the relatively successful surveillance program and the gradual decrease in introduction of new cracks in later cycles leads to a decrease of tube rupture susceptibility as seen in the results shown in Fig. 7. However, based on the fraction of SG tubes plugged at the time that SLB-induced ruptures are predicted to occur, the SGs would probably have been replaced before a tube rupture is likely. Although ruptures have occurred in SG tubes in U.S. PWRs, they have not necessarily ruptured as a result of SCC. The critical crack length at which a tube is vulnerable to rupture in the event of a steamline break is significantly longer than the length of the crack at which the crack penetrates the wall. Data are available on the reliability of eddy current testing for cracks up to the through-wall condition but not for long through-wall cracks. The assumptions made in this study about the detection of cracks after wall penetration had a significant impact on the assessed probability of SG tube rupture.

Fig. 8 illustrates the time-dependent risk of a SLB/MSIV/SGTR event. This risk is determined by assessing the probability of a SLB occurring during a cycle in which there are SG tubes with cracks that exceed the critical length of 33.0 mm (see Section 3.3.2). In this state, tubes with these cracks are vulnerable to failure if the secondary side of the SG is depressurized. After ten years of operation, the probability of a SLB is the SLB frequency times the cycle length (two years). The associated probability of a MSIV failure is four times the conditional probability of MSIV failure per demand, because the plant has four SGs protected by four MSIVs. As pointed out in Section 3.3.2, the critical crack length at which a tube becomes susceptible to failure resulting from depressurization of the SG (33.0 mm) is shorter than the critical length for a spontaneous rupture (62.5 mm). Naturally, the time at which the scenario probability becomes non-zero occurs earlier than the time at which a spontaneous SGTR becomes non-zero. More specifically, this time for SLB-induced rupture occurs during the eighth cycle, while for the spontaneous SGTR this vulnerability is activated during the ninth

cycle. The time-dependent core damage frequency for this scenario peaks at 20 years. The scenarios identified as melt scenarios in Fig. 2 involve containment bypass and the potential for a large release of radioactive material to the environment. Scenarios 5 through 8 in Fig.2 would contribute to the LERF. Since a quantitative analysis of this contribution requires a Level 3 PRA with site specific conditions, such a quantitative analysis is not considered in this study.

These results should not be interpreted as indicating a significant risk associated with the steam-line break induced rupture of SG tubes. The scenario was chosen as one that involved both degradation mechanisms leading to initiating events and degradation mechanisms affecting the response of safety-related systems, structures and components. The assumptions listed in Section 3.4 and the likelihood that the plant would have changed SGs before these conditions were reached must be recognized

4. CONCLUSIONS

The results of the case study performed illustrate how the surveillance data on passive components can be incorporated into the condition-dependent PRA, at least for a degradation mechanism for which there has been significant experience. In a real-world application, the level of effort required would be substantially greater than in our case study, because it would be necessary to identify a number of vulnerable locations in the plant and to consider alternative degradation mechanisms. These parameters would then have to be analyzed using simulations of degradation progression that would guarantee a high degree of correlation between predicted outcomes and actual plant condition. 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 600MA SG tubes was retrospective in nature. Thus, there were data available to initially tune the parameters in the degradation models, as well as some indication of those types of degradation mechanisms that would be dominant. Comparison of the results of the degradation model for the case study with actual PWR operating experience in terms of the number of tubes plugged shows that the degradation model appears slightly conservative, possibly due to the assumptions listed in Section 3.3.3. If we were to do a prospective analysis for the future risk associated with Alloy 690 SG tubes, we would need to consider another model that considers this specific material. 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 as shown in Fig.1. It is at the stage of comparing surveillance results with projections made with the degradation model that significant deviations from the functional form of the degradation model could be identified leading to the recognition of new degradation phenomena.

An important consideration highlighted by the case study is the need to have a quantitative understanding of the probability of failure of surveillance tools (see Sections 2.2, 3.3.2 and 3.3.3). This is an area in which more research is needed.

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**Table 1. Top Events and Failure Probabilities
for Steam Line Break Outside Containment (SLB) [33]***

Top Event Acronym	Top Event	Probability of Failure on Demand
K	Reactor trip	1.8E-4
M	MSIV failure	1E-4
TR	SG tube rupture given pressure gradient	0 or 1 Dependent on State of SG Tubes (See Section 3.3.2)
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

*The frequency of SLB is 0 up to Cycle 10 and 1.88E-3/yr after that (See Section 3.2.2).

Table 2. Example tube materials, geometry, operating conditions, and crack morphology

Material Parameters	
Material	Alloy 600MA
Ultimate tensile strength (UTS), σ_U	713 MPa [42]
Yield strength (YS), σ_Y	362 MPa [42]
Flow stress, σ_F	645 MPa $[0.6 \cdot (\sigma_Y + \sigma_U)]$
Young's modulus of elasticity, E	200 GPa [43]
Poisson ratio, ν	0.31
SG/Tube Parameters	
Tube outside diameter	22.23 mm [39]
Tube thickness	1.27 mm [39]
Number of tubes per SG	3592
Number of SGs	4
Operating Conditions	
Primary side pressure	17.13 Mpa
Secondary side pressure	7.48 Mpa
Crack Morphology	
Type	Axial
Location	Roll transition zone
Depth-to-length ratio	1:3
Initial crack length	0.1 mm

Figure 1. Evolution of Dynamic, Condition Dependent PRA [44]

Figure 2. SLB-induced SGTR Event Tree

Figure 2. SLB-induced SGTR Event Tree. Top branches indicate desired behavior (Success)

Figure 3. Crack Initiation Time [35]

Figure 4. Measured Crack Length Distribution at Ringhals Plant (adapted from [36])

Figure 5. Crack Length as a Function of Time for Representative Groups

Figure 6. Number of Plugged Tubes Per SG as a Function of Cycle

Figure 7. Probability of at least one SG tube rupture following a SLB per SG

Figure 8. Core Damage Frequency for SLB/MSIV/SGTR Events

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