Annals of Nuclear Energy 87 (2016) 696–706

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Implementation of condition-dependent probabilistic risk assessment using surveillance data on passive components

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ARTICLE INFO

Article history:
Received 19 January 2015
Received in revised form 25 July 2015
Accepted 27 July 2015
Available online 24 August 2015

Keywords:
Probabilistic risk/safety assessment
Plant condition monitoring
Dynamic risk assessment
Condition-dependent PRA

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 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.

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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 (Andresen et al., 2007). 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 (Hakobyan et al., 2008). Considerable research has been performed to explore the benefits of dynamic event trees (DETs) (Aldemir, 2013) 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 (Fletcher and Schultz, 1992), MELCOR (Gauntt, 2006)). 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 (Chang and Mosleh, 1999), ADAPT (Catalyurek et al., 2010), and MCDET (Kloos and Peschke, 2006) have been developed to implement these approaches. In the Risk Informed Safety Margins Characterization Program

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http://dx.doi.org/10.1016/j.anucene.2015.07.035
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(Griffit et al., 2012), 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 (Griffit et al., 2012). 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 Lewandowski et al. (2014) 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 (Andresen et al., 2007) 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 (Nuclear Energy Agency Committee on the Safety of Nuclear Installations, 2004). 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. Ramuhalli et al., 2013).

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 (Andresen et al., 2007). 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 (Dooley and Chexal, 1999; Ahmed, 2012). 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 (Spencer et al., 2006).

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 (U.S. Nuclear Regulatory Commision, 2000).

2.2. The framework (Lewandowski et al., 2014)

The modes and rates of degradation processes typically depend on the time-dependent thermal–hydraulic and stress environment to which the SSCs are exposed. Fig. 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*

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