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Condition-based probabilistic safety assessment of a spontaneous steam generator tube rupture accident scenario

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ABSTRACT

Condition-Based Probabilistic Safety Assessment (CB-PSA) makes use of the information made available during operation by sensors and/or inspections on the state of components and systems. This allows specializing the PSA to the conditions of the components and systems, reducing the uncertainty on the risk measures quantified.

In this paper, we demonstrate the CB-PSA with reference to a spontaneous Steam Generator Tube Rupture (SGTR) accident scenario in a Pressurized Water Reactor (PWR). Results show that the updated risk measures are capable of reflecting the actual state of the SG in the tailored risk evaluation.

1. Introduction

Probabilistic Safety Assessment (PSA) is a framework that systematizes the available knowledge for analyzing design and operational vulnerabilities of complex systems and quantifying the related risk measures (IAEA-TECDOC-737, 1994; IAEA-TECDOC-1106, 1999; Nakai and Kani, 1991). In Nuclear Power Plants (NPPs) these are, for example, Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) (Zubair et al., 2011). PSA involves system accident analysis by techniques like Event Tree Analysis (ETA) and Fault Tree Analyses (FTA) to define accident sequences and calculate the frequency of the system end states (for example, Core Damage) reached along the accident scenarios.

As knowledge on the components states changes during the system life, PSA results need to be updated by embedding failure data, physical knowledge and monitored data on the conditions of the components. For example, in the Living PSA (LPSA) paradigm the PSA is updated to reflect plant changes and embed field failure data (IAEA-TECDOC-1106, 1999). LPSA is a plant specific PSA that can be updated or modified to reflect the plant changes during the lifetime (Johanson and Holmberg, 1994). Changes can be physical (resulting from plant modifications, etc.), operational (resulting from enhanced procedures, etc.), organizational, but can be also changes in knowledge due to the acquisition of operational experience, field failure data, etc. The updated LPSA, then, reflects the current design and operational state of the system, and is documented in a way that each aspect of the model can be directly related to existing plant information, plant documentation or analysts assumption (IAEA-TECDOC-1106, 1999; IAEA, 2008).

In this paper, we extend and advance LPSA by proposing a Condition-Based PSA (CB-PSA) framework. This novel concept is here presented for the first time, extending the capability of LPSA by incorporating knowledge on the state of the components, as estimated from monitored data. The development and renovation of Non-Destructive Examination (NDE) technology has also improved the quantity and quality of the available data (Obrutsky et al., 2014), boosting the development of techniques aimed to process those data and information to increase the capability of PSA and LPSA to provide actualized, tailored and robust risk measures. CB-PSA actualizes the risk measures to the current state of the components, based on the information on the components states made available by monitoring or inspection. This entails integrating PSA techniques like ETA and FTA with condition monitoring techniques (Varde and Pecht, 2015; IAEA-TECDOC-1106, 1999; Aldemir, 2013; Poghosyan and Amirjanyan, 2015; Kim et al., 2015; Hines et al., 2008; IAEA, 2013) that provide estimates of components states and associated uncertainties, thus reducing conservativism and uncertainty on the risk measured quantified (Zio, 2016). The empirical distributions based on field failure data of LPSA are replaced by condition-based distributions obtained by the integration of physics-based knowledge and monitoring data on the ongoing degradation mechanisms and aging phenomena affecting the components, as well on environmental and operational conditions. Fig. 1 gives the idea by sketching the failure probability given by conventional PSA (dotted line), the updated failure probability computed by LPSA (continuous line) and the condition-based failure

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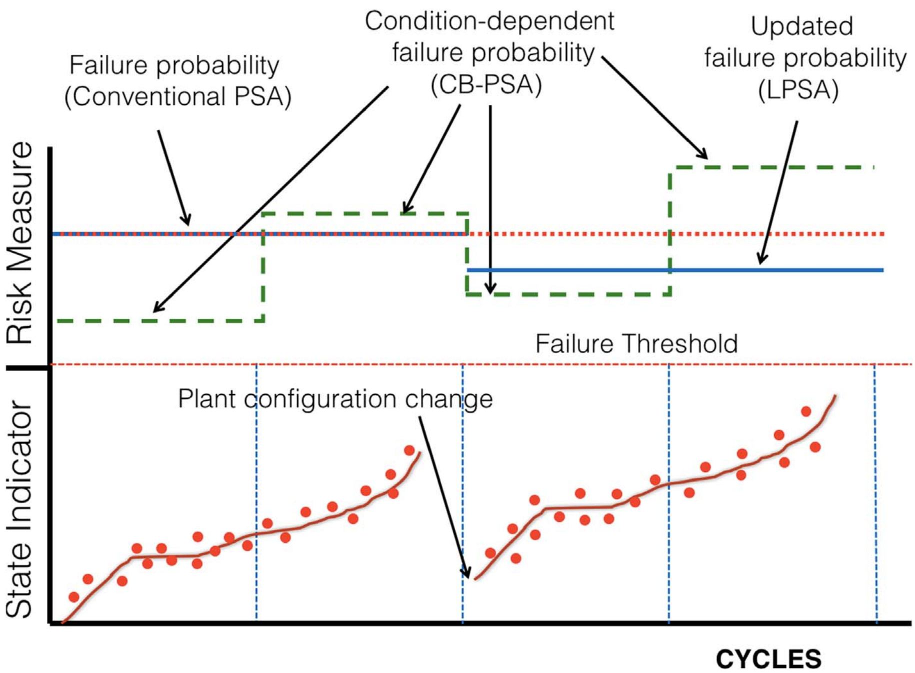

Fig. 1. Comparison between failure probability provided by conventional PSA and condition-based failure probability provided by CB-PSA.

probability provided by CB-PSA (dashed line). Conventional PSA estimates the failure probability based on the knowledge and information available before operation, neglecting future time-dependent variations, e.g. due aging, failure and maintenance; LPSA updates conventional PSA to reflect changes in the plant configuration; CB-PSA uses condition-monitoring data for updating the failure probability at each cycle time.

The remainder of this paper is as follows: in Section 2, the case study of a spontaneous Steam Generator Tube Rupture (SGTR) in a pressurized water reactor (PWR) is considered and the models used to implement the CB-PSA are described. In Section 3, the CB-PSA framework is presented along with a plugging optimization methodology for limiting the occurrences of a spontaneous SGTR accident event. Section 4 draws some conclusions.

2. The spontaneous SGTR accident scenario

SGTR can be either an induced or a spontaneous phenomenon. An induced SGTR consist in the break of one or more SG tubes that is triggered by other internal events, such as a Steam Line Break (SLB), whereas a spontaneous SGTR is not (NUREG/CR-6365 INEL-95/0383, 1996).

Fig. 2 shows the simplified ET that follows to a spontaneous SGTR Initiating Event (IE). The frequencies of the events along the sequences in the ET are estimated from statistical analysis of reliability data and expert judgement, if needed (Kim et al., 2015).

The frequency f_{SGTR} of the spontaneous SGTR IE of Fig. 2 (which is used within the conventional PSA, performed for the Safety Assessment Review (SAR) that has to be submitted to the regulatory authority for licensing) is given by Eq. (1):

$$f_{SGTR} = \frac{N + 1/2}{T} \tag{1}$$

where N is the number of SGTR occurrences in T years of similar NPP operations (for example, $N = 3$ in $T = 499$ years as reported in (Sattison and Hall, 1990), resulting in f_{SGTR} = 7.0E-03 per year (Kim et al., 2015)).

Assuming the failure on demand probability of the operator depressurization *OD*, the failure probability of refill of the storage tank *RWST* and the failure probability of the reactor safety system *RSC* equal to 1.8E-4, 2.4E-8, 5.6E-5, respectively (Lewandowski, 2013), the CDF is equal to 3.92E-7 per year.

In this work, the spontaneous SGTR is analyzed within a condition-based PSA. A model for the onset, formation and propagation of spontaneous cracks in the SG is used to update the probability of the SGTR IE throughout the system lifetime and the CDF is updated by the analysis of the current state of the plant.

2.1. The steam generator

LPSA and CB-PSA are plant specific PSA. To show the capability of the proposed CB-PSA approach to follow the specificities of the system under analysis and to tailor its specific operative conditions, we focus on the SG of the Zion PWR NPP, equipped with a recirculating SG of 3.6 m and 21 m of diameter and height, respectively, 800 t of weight, a bundle of 3592 inverted U tubes with an outside diameter of 22.23 mm and a wall thickness of 1.27 mm (Lewandowski, 2013). The primary loop nominal pressure is 15.2 MPa, while the secondary loop nominal pressure is 6.9 MPa. The hot leg nominal temperature is 330 °C, while the nominal cold leg temperature is 288 °C. A detailed list of Zion NPP parameters values is given in Table 1.

2.2. The spontaneous SGTR model

Degradation of SG tubes largely impact NPPs operation. The most common form of degradation leading to failure is Stress Corrosion Cracking (SCC) that accounts for 60% to 80% of all tube defects requiring plugging. Fretting and pitting collectively account for another 15% to 20%, whereas the remaining failures are due to mechanical damage, wastage, denting, and fatigue cracking (Wade, 1995; Chatterjee and Modarres, 2011). For this reason, without loss of generality, the spontaneous SGTR (with the associated tube cracking) is here considered to be only due to SCC (Cizelj and Mavko, 1995) and we do not consider the management of leaked tubes but only the management of tube ruptures, for simplicity.

The tube cracking process can be divided into onset, formation and propagation of cracks inside the tube well. The crack onset (i.e., the generation of microcracks inside the tube bundle) is modeled relying on the actual data collected in the Zion Plant (see (Lewandowski et al, 2016), for further details). Specifically, after 4 years (i.e., 2 refueling