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Quantification of initiating event frequencies and component reliability data in level 1 probabilistic safety assessment at Puspati TRIGA research reactor

M. Maskin [a,b], A. Hassan [c], F.C. Brayon [d], P.T. Phongsakorn [b], M.F. Zakaria [b], Z. Ramli [d], F. Mohamed [a,*]

[a] Faculty of Science and Technology, University Kebangsaan Malaysia, Selangor, Malaysia
[b] Reactor Technology Center, Malaysian Nuclear Agency, Selangor, Malaysia
[c] Faculty of Applied Sciences and Technology, Universiti Tun Hussein Onn Malaysia, Johor, Malaysia
[d] Atomic Energy Licensing Board, Selangor, Malaysia

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ABSTRACT

Prior to developing level-I probabilistic safety assessment in Puspati TRIGA Reactor (RTP), collecting and analyzing data is the cornerstone of the probabilistic safety assessment processes. This paper provides on how data quantification is made available for both initial initiating events frequency and components reliability. Taking into consideration on the availability of data, it was decided that generic data to be used in combination with plant-specific data. Methods for estimating the parameters used in quantifying is also presented in this paper. This includes, data treatment by applying Bayesian updates in forming a better refine posterior data. It was found out that, result with the presence of plant-specific data history tends to have high value, loss of offsite power (LOOP) with 1.27E-03. Meanwhile, for other IEs that used generic data, shown lower result with maximum value of 2.68E-06 (loss of flow accident at secondary pump: LOSC-P). As for components data, basic event, No signal from fission chamber 1 (fail to function) is the highest value with 4.07E-01, followed with Manual valve SV1A fail to close (2.75E-01). This publication is perhaps one of the first publically available document providing information on generic and plant specific data for initiating event and components reliability parameters that were used in a Level-1 PSA study for a nuclear research reactor. In addition, this information can be used by PSA technical staff in how to analyze and treat initial incident event data, including, type and what data need to be collected.

1. Introduction

Probabilistic safety assessment (PSA) technique is increasingly being used in many countries operating research reactor as an assessment tool of risk associated with a broad range of potential accident scenarios, identify adverse effects of various risk contributors (e.g. equipment failures, human errors, etc.), and determines the measures for further enhancement of plant safety by incorporating the risk insights in the decision-making process (IAEA, 1987; IAEA, 2010; Brayon et al., 2014; IAEA, 2001).

PSA is usually developed based on logical and systematic approach that makes use of realistic assessments of equipments and plant personnel performance as a basis for calculations of risk. This in principle has the potential to produce an understanding of the inherent risk of the operating plant over a much wider range of conditions than the traditional deterministic methods which generally define what is assumed to be a bounding set of fault conditions (IAEA, 1200; Barón, 2010; IAEA, 1989). Furthermore, the adoption of conservative assumptions relating to plant and system performance is an accepted approach to address uncertainty when performing deterministic analyses. The combination of considering a limited number of faults and a conservative approach to the analysis of each fault can produce inappropriate, or worse, misleading insights, and therefore decisions solely based on these deterministic types of analyses might not always be the most appropriate way for reducing plant risk (IAEA, 1200).

In order to estimate the frequencies of occurrence for accident sequences analyzed in PSA, three basic input parameters must be quantified: (a) initiating events (IEs) frequency; (b) component

* Corresponding author.
E-mail addresses: mazleha@nm.gov.my (M. Maskin), hassan@uthm.edu.my (A. Hassan), fedrick@aelb.gov.my (F.C. Brayon), phongsakorn@nm.gov.my (P.T. Phongsakorn), fazli@nm.gov.my (M.F. Zakaria), zulfadli@aelb.gov.my (Z. Ramli), faizalm@ukm.edu.my (F. Mohamed).

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failure probabilities; and (c) component unavailabilities. However, component unavailabilities is not cover in the scope of present study.

A level I probability safety assessment (PSA) study was conducted at Puspati TRIGA Mark II research reactor (RTP). This project which started in 2014 was monitored and mentored by the IAEA safety assessment division under a project named COMPASS-M (**Com**petence for **P**robabilistic **As**sessment of **S**afety in **M**alaysia). With 1 MW(th) power, RTP has been operating for 35 years since the achievement of its first criticality on June 28, 1982. Using Americium-Beryllium as the neutron source, RTP reactor power is controlled by levelling up or down 4 control rods, namely: transient (an air follower control rod), and 3 fuel follower control rods (regulating, safety and shim rod). Heat generated in the reactor core are transferred to the water coolant by natural convection. The heat is then, subsequently removed by pumping the water via the primary cooling system through a plate-type heat exchanger (HE) and then via the secondary cooling system into the cooling tower for the final heat dissipation.

The main objective of this paper is to provide on how the frequencies of the occurrence of incident events and parameter estimations for components failure are collected and estimated. In particular, this paper is intended to serve as a guidance in collecting and calculating incident and component event datas for a nuclear research reactor as the prior steps in developing level-I PSA for internal initiating events. Criteria and scope of study to determine events to be included are: (i) internal initiating events; (ii) based on selected postulated identified IEs in the previous work by Maskin et al. (2016); (iii) unplanned reactor trip (not a scheduled reactor trip on daily operation schedule); and (iv) events that occurred during calendar years 1985 through 2015. Data on operator actions (Mohamed et al., 2015; Hassan et al., 2017) is not taken into account in this paper.

Section 1 of this paper provides the introduction, including the purpose and scope of the current study and Section 2 addresses the methodology of the current study. Section 3 describes the sources and results that are to be used, and how the data were organized and treated. Sections 4 and 5 contains the conclusion of the study and the authors' acknowledgment, respectively.

2. Methodology

2.1. Data source

In quantifying the two parameters of interest mentioned in Section 1, which are: (a) initiating events (IEs) frequency; and (b) component failure probabilities, two sources of data are involved: (i) plant specific data (PsD); and (ii) generic data.

Fig. 1 shows the hierarchy in identifying data type and sources. PsD or RTP raw data is collected and used for both IEs and components event data. To this end, available operational log books and maintenance reports dated from 1985 through 2015 are examined. As no formal systems for recording abnormal events and component failure data had been in place, the collection of data from both log books and reports are conducted manually and stored in Microsoft Excel.

Generic data for IEs are taken from the IAEA incident reporting system for research reactor (IRSRR) database. It is a system that collects, analyse, maintain and disseminate information received from participating Member States of the IAEA on unusual events that have occurred at research reactors, including reports that occurred before the IRSRR came into effect (IAEA, 2014). The reported incidents from the database are screened based on few criteria in search for the IE that is in line with type of list of IEs postulated in the RTP's PSA (Maskin et al., 2016). In this step, identifying the IE is based on the consideration of the detail reported information provided in the IRSRR by expert judgement approach.

Meanwhile, generic data for components event data are taken from TECDOC-930 (IAEA, 1987). The selection of information that is to be extracted from TECDOC-930 is based on the degree of similarity of the reactor to RTP. For example, Bandung research reactor from Indonesia (IN-B) is chosen to be the first choice in the selection, due to the fact that it is the same type TRIGA Mark II reactor. In the case when the analyzed components are not available from the IN-B, then, other similar reactor will be chosen. This process is iterated until the entire data for the RTP specific component is available. All data are recorded and analysed using Microsoft Excel.

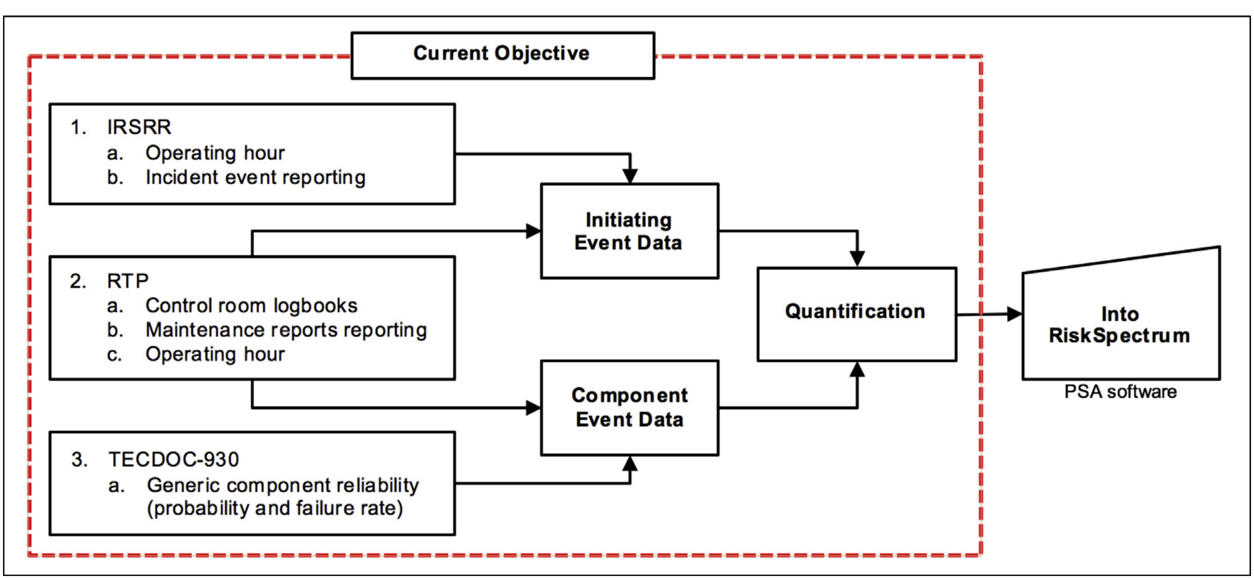

Fig. 1. Identification and quantification of data sources (IRSRR, RTP and TECDOC-930) in estimating initiating and component event data as PSA model input.