Building Competence for Safety Assessment of Nuclear Installations: Applying IAEA's Safety Guide for the Development of a Level 1 Probabilistic Safety Assessment for the TRIGA Research Reactor in Malaysia

F.C. Brayon*[a], M. Mazleha[b], P. Prak Tom[b], A.H.S Mohd Sarif[c], Z. Ramli[a], F. Zakaria[b], F. Mohamed[c], Abid Aslam[d], A. Lyubarskiy[e], I.Kuzmina[e], P.Hughes[e] , A.Ulses[e]

[a]Atomic Energy Licensing Board, Selangor, Malaysia
[b]Malaysia Nuclear Agency, MOSTI, Selangor, Malaysia
[c]Universiti Kebangsaan Malaysia, Selangor, Malaysia
[d]Pakistan Nuclear Regulatory Authority, Pakistan
[e]International Atomic Energy Agency, Vienna, Austria

Abstract: In 2010, the International Atomic Energy Agency (IAEA) published its Safety Guide on Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (i.e. SSG-3). Although it was aimed at covering state-of-the-art recommendations for the development of PSA of Nuclear Power Plants (NPP), the guidance was deemed to be applicable to other nuclear installations as well. In order to get insights regarding the applicability of SSG-3 for a research reactor, and at the same time to achieve the goal of building competence and capacity for PSA in Malaysia, in December 2012, the IAEA started an extra budgetary project entitled "Applying PSA to Existing Facilities to Develop Transferable Skills in the Use of PSA to Evaluate NPP Safety." This project has been funded by Norway. The facility selected for the PSA study was the TRIGA Puspati Research Reactor (1 MW) which has been in operation in Malaysia since 1982. All major PSA tasks have been performed in accordance with the recommendations provided in SSG-3 (e.g. initiating events analysis, systems analysis, component data evaluation, human reliability analysis, etc). The design specifics of the research reactor under consideration have been addressed in the PSA model (e.g. four end states have been defined, detailed consideration of initiating events induced by human errors, several operational states, etc.). The paper provides an overview of the methodology applied and discusses specific features of PSA tasks for the research reactor. Preliminary results and insights obtained are presented. In addition, insights for guidance in developing a research reactor PSA are highlighted in the paper.

Keywords: IAEA SSG-3, *PSA, TRIGA, research reactor*

1. INTRODUCTION

The International Atomic Energy Agency (IAEA) is helping Member States wishing to employ nuclear power option to build competence and capacity for nuclear safety assessment. The Safety Assessment Section (SAS) of the Division of Safety of Nuclear Installations (NSNI) is conducting many activities aimed at providing advanced training for state-of-the-art safety assessment techniques including Probabilistic Safety Assessment (PSA). For the latter, IAEA Safety Standards on PSA [1, 2] and other IAEA publications [3, 4] have been widely used as a basis for the training.

In March 2012, IAEA started an extra budgetary project (EBP) sponsored by Norway (NOKEBP) to build competence and capacity for nuclear safety assessment in Vietnam and Malaysia. The training comprised theoretical and extensive practical parts. In the practical part, simulation of PSA performance for a simplified Nuclear Power Plant (NPP) was performed; the training approach is described in Reference [5].

*fedrick@aelb.gov.my

After completion of the training sessions, Malaysia applied to IAEA for a continuation of the training to further enhance practical skills in PSA. Responding to the request, under the same EBP project sponsored by Norway, in December 2012 IAEA launched a project entitled "Applying PSA to Existing Facilities to Develop Transferable Skills in the Use of PSA to Evaluate NPP Safety". The short name of the project is COMPASS-M (Competence for Probabilistic Assessment of Safety in Malaysia). The facility selected for the PSA study is the TRIGA Puspati Research Reactor (RTP) which has been in operation since 1982. Several Malaysian organizations cooperate under the COMPASS-M project, namely Atomic Energy Licensing Board (AELB), Malaysia Nuclear Agency (MNA), and National University in Malaysia (UKM). This project is scheduled to be completed in mid-2014.

2. SCOPE AND OBJECTIVE

The objective of COMPASS-M is to master the PSA technology for safety assessment of NPPs through practical application of relevant recent IAEA Safety Standard on PSA [2] to develop a PSA model for the research reactor in Malaysia.

Three main outputs are expected from the project; these are the following: (a) state-of-the-art competence for PSA in Malaysia (applicable to NPPs and other nuclear installations); (b) a PSA model and a PSA report for the operating research reactor in Malaysia; and (c) insights for future IAEA's publications on PSA for research reactors.

The results of the PSA study will be used as a complementary information source for further safety improvement of the RTP. The latter may include, for instance, evaluation of the design and possible modifications, establishing effective maintenance programs, and training of operating and maintenance personnel.

The scope of this study covers Level 1 PSA for internal initiating events caused by random component failures and human errors. The analysis is performed for full power and daily shutdown operational conditions; the shutdown for annual maintenance is not included in the PSA scope. The radioactivity source considered in the study is the reactor core only.

3. PROJECT MANAGEMENT AND ORGANIZATION PROCEDURE

The project management approach envisages three main mechanisms: (a) home work by the Malaysian PSA team; (b) quarterly Technical Review Meetings (TRMs); and (c) semi-annual Steering Committee Meetings (SCMs). The objective of TRMs is to provide guidance for selected PSA tasks and specify the scope of the home work to be done by the Malaysian PSA team until the next TRM. The role of SCMs is to provide an oversight of the project conduct and help in resolving emergent issues, e.g. access to information and allocation of resources. Managers and supervisors from IAEA and Malaysia are the members of the Steering Committee.

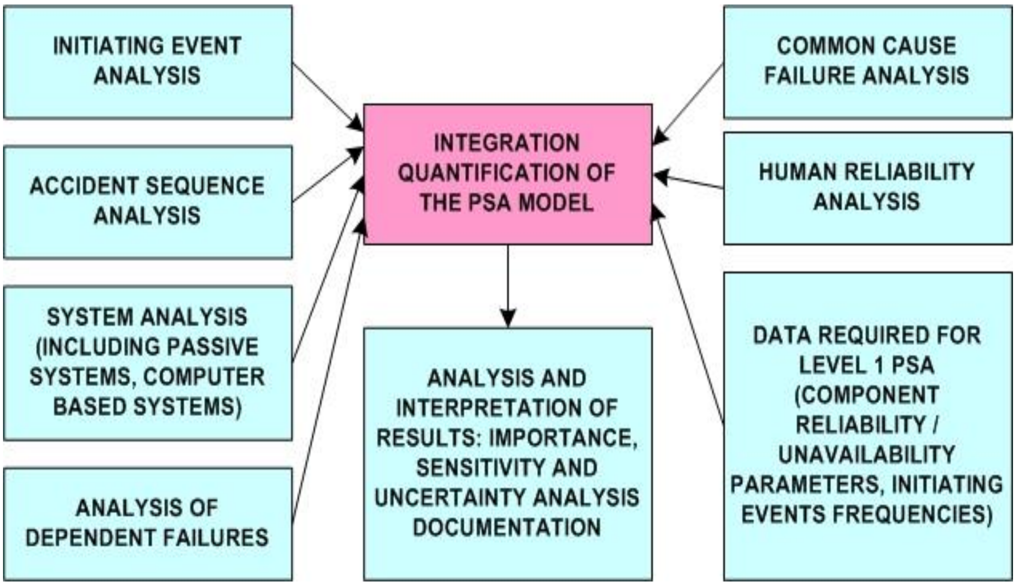
The PSA team comprises six specialists from AELB, MNA, and UKM. The team is being led by the Technical Project Manager from AELB and the Deputy Project Manager from MNA. The PSA tasks were distributed to the PSA team based on the composition of the PSA tasks shown in Figure 1 copied from the IAEA Safety Guide on Level 1 PSA [2].

4. BRIEF DESCRIPTION OF THE TRIGA RESEARCH REACTOR IN MALAYSIA

4.1. Historical Review

The Malaysian TRIGA Mark II research reactor was supplied and constructed by General Atomic Co. Ltd. San Diego, California, United States of America and built in 1981. It came into operation in 1982 and reached the first criticality on 28 June 1982 at nominal power of 1000 kW.

Figure 1. Tasks of Level 1 PSA for Internal Initiating Events from IAEA Safety Guide SSG-3



RTP is owned by the Government of Malaysia and operated by MNA, a government organization under the administration of the Ministry of Science, Technology and Innovation (MOSTI). The facility is located at Bandar Baru Bangi Selangor, around 50 km from Kuala Lumpur, the capital city of Malaysia.

4.2. General Description of the RTP Facility

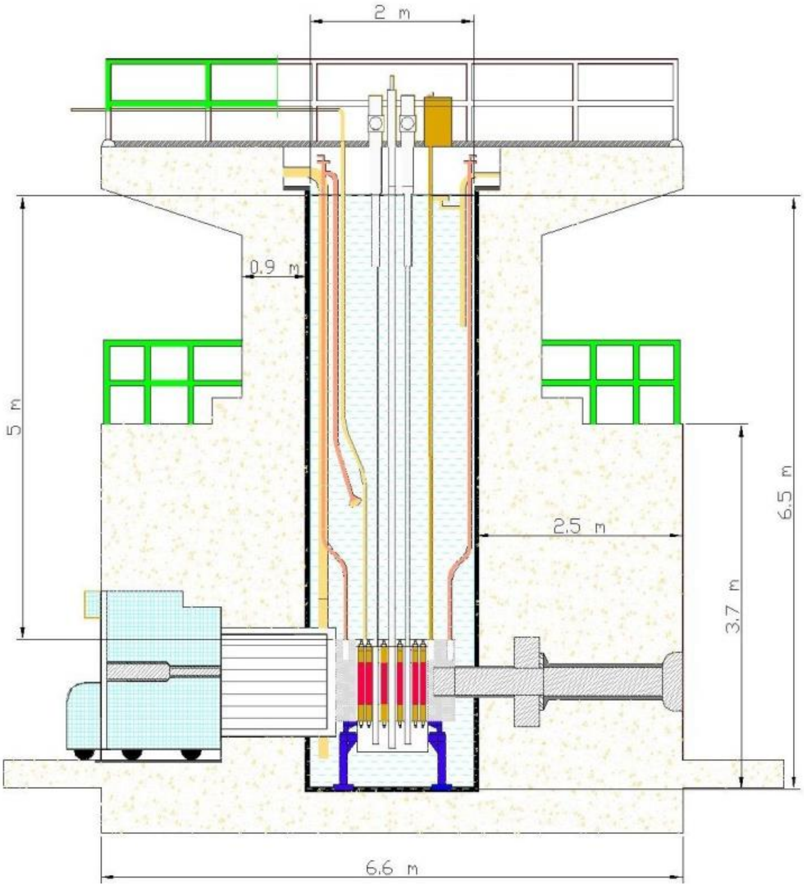
RTP was designed to effectively implement the various fields of basic nuclear research and education. It incorporates facilities for advanced neutron and gamma radiation studies as well as isotope production, sample activation and student training.

RTP is an open pool-type reactor with its core and reflector assemble immersed at the bottom of a 2-m diameter aluminium tank as shown in Figure 2. The reactor core and experimental facilities are surrounded by a concrete shield structure. Approximately 5 m of water above the core provides vertical shielding. The core is shielded radially by high density concrete, water and graphite reflector.

The reactor core utilizes a solid, homogenous fuel element developed by General Atomic, USA, in which the zirconium-hydride ($ZrH_{1.6}$) moderator is homogenously combined with low-enriched uranium (LEU). The unique feature of these fuel-moderator elements is the prompt negative temperature coefficient of reactivity, which equips the TRIGA reactor with its built -in safety feature by automatically limiting the reactor power to a safe level in the event of a power excursion. The reactor core consists of a circular array of cylindrical fuel-moderator elements and graphite (dummy) elements. The fuel elements have 8.9 cm long graphite end sections that formed the top and bottom reflectors. About one-third of the core volume is occupied by water. 25.4 cm thick of graphite radial reflector surrounds the core and the entire assembly is supported on an aluminium framework at the bottom of the tank. Physical access to the core and observation of it are possible at all times through the vertical water shield.

RTP has four beam ports extend from the reactor assembly through the water and concrete to the outer face of the shield structure. A rotary specimen rack in an annular well in the top of the graphite reflector provides for the large-scale production of radioisotopes and also for the activation and irradiation of small specimen. All 40 positions in this rack are exposed to neutron fluxes of comparable intensity. The RTP is also equipped with a central thimble for conducting experiments or irradiating small samples in the core at the point of maximum flux. Experimental tubes can easily be installed in the core region to provide additional facilities for high-level irradiation or in-core experiments. A high-speed pneumatic transfer system permits the use of extremely short-lived isotopes. The in-core terminus of this system is located in the outer ring of fuel elements positions, a region of high neutron flux.

Figure 2. Cross-cut View of RTP



The power level of the RTP is controlled using four control rods: a regulating rod, a shim rod, a safety rod and a transient rod. In accordance to the safety concern, transient tests at General Atomic have proved conclusively that the large prompt negative temperature coefficient of the fuel moderator material provides a high degree of self-regulation without the assistance of external control devices.

The water-cooling and purification systems maintain low water conductivity, remove impurities, maintained the optical clarity of the water and dissipating the reactor heat. They consist of a water surface skimmer, pump, filter, demineralizer, heat exchanger unit, associated piping and valves and miscellaneous instrumentation.

4.3. Experimental Program

As mentioned above, the reactor is used as a neutron source for variety of experiments using the experimental facilities, i.e. the vertical irradiation channel, in-core irradiation facility and the horizontal beam ports of the reactor. The experimental program [7] conducted at the RTP facility includes radioisotope and radiopharmaceutical production, analytical analysis, nuclear physics research as well as reactor physics and thermal-hydraulic research.

4.4 Safety Features

The RTP has been designed to prevent accidents and to mitigate potential accident consequences, as well as to protect the site personnel, the general public and the environment from the radiation hazards. The safety systems of the RTP are the reactor control and protection systems for the reactor scram signals, and for reactor protection against occurrence of reactor overpower, reactor pool with primary coolant for core cooling by natural convection, uninterrupted electric power supply system for uninterrupted power supply of the most safety-significant systems and reactor hall with special ventilation system for control of radioactive releases into the environment. The safety features ensure the safe shutdown of the reactor, removal of heat from the core and limiting the consequences of anticipated operational occurrences and accident conditions.

5. OVERVIEW OF THE PSA DEVELOPMENT PROCESS

5.1. PSA Procedures

For each PSA task shown in Figure 1, procedures describing the work to be done and approaches to be used have been developed taking into account the specificity of the RTP. IAEA Safety Guide SSG-3 [2] and TECDOC-1511 [4] on PSA quality for applications served as a basis for the procedures. The specific features of the RTP have been addressed. The procedures developed serve as reference documents throughout the course of the PSA development.

It is expected that these procedures will be a useful example for any parties interested in performing a Level 1 PSA study for a TRIGA research reactor.

5.2. Plant Familiarization and Information Collection Procedure

A walkdown of the RTP was performed in the beginning of the project. Findings of the walkdown accompanied by pictures were documented in a formal technical report containing 27 walkdown sheets, which served as a valuable information source to verify as-build as-operated RTP conditions. All relevant drawings and layout schemes were collected as well. In addition, a considerable amount of data on plant operational history, incidence data, and component failures has been collected.

5.3. Initiating Event (IE) Analysis

The initiating events analyses is the starting point of the Level 1 PSA aimed at identification of a set of initiating events that require mitigation to prevent core degradation and then classification of these initiating events in specific groups for further analysis.

The objective of initiating event analysis is to ensure that the set of initiating events identified is as comprehensive as possible to provide confidence in the completeness of the probabilistic safety assessment.

Following the recommendations of SSG-3 [2] five approaches were utilized in this analysis: (a) Master Logic Diagram (MLD); (b) Failure Mode and Effect Analysis (FMEA); (c) analysis of the list of postulated initiating events (PIE) considered in the Safety Analysis Report (SAR) [6]; (d) analysis of generic lists of IEs presented in TECDOC-400 [8]; and (e) analysis of operating experience (specific and generic). As a result of these analyses, more than sixty IEs for the RTP have been identified; these events have been further grouped into eight IE groups:

(1) Small break loss of coolant accident in the confinement (SLOCA-C);

(2) Large break loss of coolant accident in the confinement (LLOCA-C);

(3) Loss of flow accident due to pump failure (LOFA-P);

(4) Loss of flow accident due blockage at heat-exchanger plate (LOFA-B);

(5) Loss of offsite power (LOOP);

(6) Loss of flow accident due to secondary pump failure (LOSC-P)

(7) Reactivity insertion accident (RIA);

(8) Loss of instrument and control (LOI&C).

5.4. Accident Sequence (AS) Modelling

Definition of End States

Five end states have been defined in the PSA based on the degree of possible degradation of the reactor core and the potential for direct releases to the environment. These end states are listed in Table 1.

Table 1. Definition of End States

Core Damage State	Description	Conditions
1-FDRF	Fuel degradation during refilling	RPS failure, partial core uncovery, minor degradation during core refilling by external water
2-FDEA	Fuel degradation due to exposure to the air	RPS success, heat remover fail, complete core uncovery. Minor degradation to the fuel is expected (core uncover with successful reactor SCRAM).
3-CDG	Core (fuel) degradation	RPS failure, rapid core uncovery.Major degradation to the fuel is expected (core uncover with failure in reactor SCRAM)
RPWC	Release of pool water outside confinement	Release of pool water outside confinement in condition of degraded core. Applicable to LOCA events with non OK end state
OK	No degradation of the fuel	No degradation of the fuel

Accident Sequence Models

In modelling the accident sequences, an approach of small event tree (ET) and large fault tree (FT) was used. The accident sequence models have been developed for all seven IEs groups identified in the IEs analysis task.

In the process of model development, the information from RTP operation manuals and available procedures and SAR, as well as the information obtained from intensive discussions with senior reactor operator (SRO) and senior reactor staff has been extensively used. The professional PSA software RiskSpectrum [9] was used in constructing both event trees and fault trees.

Supporting Thermal Hydraulic Analysis

The SAR [6] for RTP provides limited information on the supporting thermal hydraulic analysis needed for the purpose of PSA. In order to get a realistic definition of the core degradation states and understanding of the accident progression, as well as estimates of the time windows for operator actions the supporting thermal hydraulic calculation have been performed.

A RELAP5 model was developed to simulate the entire RTP system. The model represents all reactor components of primary loops with the corresponding neutronic and thermal hydraulic characteristics.

The RELAP5 model was validated by comparing both measured and calculated data for the RTP. Comparisons include parameters of normal operation at constant power, the step reactivity transient and the control rod withdrawal transient. Good agreement was obtained for the core coolant inlet and outlet temperatures for operation at constant power, for building-up power level after reactivity insertion, and for the control rod withdrawal transient.

The developed RELAP5 model for RTP has been shown to be an appropriate model. It is being used in this PSA study to evaluate accident progression for several beyond design basis accident sequences as there was no sufficient information in the SAR to justify that the reactor will achieves a safe end state during beyond design basis accident. Specifically, the accident sequences involving LOCA and LOFA events accompanied by failure of the reactor scram function are being examined. The analysis results will be taken into account in the final models of the accident sequences.

5.5. Systems Modelling

Within the system modelling task, sixteen FTs were developed representing failure of the systems required for accident mitigation following each of the seven IEs: monitoring systems, primary and secondary cooling systems, power supply, and I&C. These FTs have been developed based on the RTP systems layout and function as well as components functions within the respective system. The system models include basic events representing components failures and latent human errors that could lead to failure of the system or post-IE human errors to initiate the system operation.

RiskSpectrum PSA software [9] was used to develop and analyse the system models. The FTs analysis provides the information on the system minimal cut sets (MCSs) for different system configurations. In addition, the importance of different basic events representing component random failures, common cause events, and human errors basic events have been obtained.

5.6. Human Reliability Analysis (HRA)

All three types of human errors usually considered in PSA (i.e. Types A, B, C) were addressed in the PSA. In modelling post-initiator human errors (Type C), most of the information and data were collected by interviewing reactor operators and reviewing available operator manuals. The SPAR-H [10] method has been used for quantifying human error probabilities (HEPs) of Type C human errors. SPAR-H worksheets allow quantifying HEPs by considering performance shaping factors (PSFs) that may increase or decrease the likelihood of errors. For the pre-initiator human errors (Type A) and initiator-type human errors (Type B), the ASEP/THERP [11] method has been used for quantifying HEPs. Identification and modelling of human actions are performed following recommendations of IAEAs' SSG-3 [2], IAEA-50-P-10 [12] and TECDOC-592 [13]. At the RTP, start-up and shut-down of the reactor are performed on a daily basis. This is taken into consideration when categorizing human errors into pre-initiator or initiator-causing categories.

The outcomes of this átask are: (a) identified human interactions (activities, experiments, manipulations, etc.) during operation of RTP; (b) quantified HEPs of Type C human errors for seven ETs taking into account different conditions, e.g. availability of alarms; (c) quantified HEPs of Type B human errors (human errors that could cause IE during operation); and (d) quantified HEPs of Type A human errors.

5.7. Data Collection and Treatment

Two types of data were collected: (a) raw specific data for RTP, and (b) generic data.

For RTP specific data, such as operational experience, data on exposure time, number of demands, initiating events frequency, number of failures and unavailability during periods of testing, maintenance and repair, were collected throughout the lifetime of the RTP. The raw data is collected from historical records, such as log books, operations reports and maintenance work records.

For RTP, the log books are generally considered to be the most accurate source of information, however the number of failures, especially on non-safety equipment, are not being recorded properly. Therefore, expert judgement by interviewing the reactor operator has been done to determine the cause of failures recorded in several reports. However, collecting data from the log books is extremely time consuming and requires detail observations as sometimes events or failures of components were not directly stated in the log books.

Maintenance work orders are considered to be the most complete records, but the reliability of the information depends on the person completing the report. In many cases these are maintenance staff themselves; therefore the accuracy of the records (which definitely not computerized) varies. In order to minimize the inaccuracy, cross-reference check was done with various files to make sure all data are consistent.

Meanwhile, for generic data, information from TECDOC-930 [14] and [16] was utilized. A comprehensive analysis of the applicability of the generic data presented in this TECDOC has been performed. The selection of generic data was based on a degree of similarity of the reactors presented in the TECDOC comparing to the RTP. The aim was to achieve the best applicable data.

For treatment of the data, mainly, the one-stage Bayesian updating approach was used for component failure rates, probabilities of failure on demand, and IE frequencies using Microsoft Excel spreadsheets. However, when plant specific statistics on failure events was sufficiently representative, only plant-specific data were used.

5.8. Dependency Analysis

The standard approaches recommended in SSG-3 [2] were utilized in the analysis and modelling of the dependencies applicable to the RTP. The dependencies addressed were the following:(a) functional dependencies, (b) physical dependencies, (c) human interaction dependencies and (d) component failure dependencies.

Functional dependencies considered in the COMPASS-M project include dependencies resulting from common support systems. These dependencies first have been identified and documented in the form of dependency matrixes and second have been directly modelled in the event trees and fault trees using RiskSpectrum features, e.g. linking fault trees for frontline systems to fault trees representing support system models. Only two support systems have been identified to be required for the inclusion in the PSA model: power supply and instrumentation and control. Other support systems (e.g. cooling systems, ventilation system) were shown to have no impact on the accident progression and operation of the frontline systems.

Physical dependencies (also referred to as **spatial interaction dependencies**) have been identified and modelled in the COMPASS-M project: dependencies due to an initiating event that disable or enable certain signals for systems actuation or alarms for operator actions. These dependencies were also modelled using applicable features of the RiskSpectrum software (e.g. boundary conditions sets and house events) that allow the model to exclude particular fault trees (or brunches of the fault trees) when it cannot be credited for the particular initiator (e.g. exclusion from the reactor scram system logic of the signal on power increase to 10% for LOCA events). The other type of physical dependencies identified in the PSA is the dependency of human actions on the available information for particular accident sequence (e.g. human error probability to perform certain action is higher when alarms designed to inform operators on the need to perform the action fails). This type of dependencies was modelled using RiskSpectrum feature “exchange event” that allows replacement of one basic event with the other depending on the specific accident sequence in the event tree. No other physical dependencies have been identified in the research reactor under consideration.

Human interaction dependencies identification has been performed under HRA task and several dependent human errors have been found and modelled in the PSA. The approach for quantification of dependent human errors was based on the techniques employed in SPAR-H method. These dependent human errors have been directly included in the PSA model.

Component failure dependencies due to errors in design, manufacture or installation are addressed by a common cause failure analysis. The following main common cause groups have been identified and modelled: pumps fail to start/fail to run; motor operated valves fail to open/ to close; thermocouples fail to operate; fission chamber fails to operate; and heat exchanger clogging.

The alpha-parameters model has been used for quantification of the CCF probabilities for all CCF groups except for CCF of control rods which use the beta-parameters. Calculation of all CCF parameters is based on NUREG/CR-6268 [15] after the applicability analysis of the component types to the RTP.

5.9. Model Integration and Quantification

The integrated PSA model for the RTP was developed using RiskSpectrum software and utilizes the small ETs – large FTs approach (fault tree linking approach). Each heading of the ETs is linked to the specific FT. For each branch in the ETs a specific end state is assigned.

The features of the RiskSpectrum software were utilized to represent specific system configurations in specified accident sequences (e.g. "boundary conditions sets", "house events"). For assigning different values for human error probabilities in different conditions, the "exchange events" feature of the RiskSpectrum was used.

The PSA model allows quantification of MCSs for specific FTs, accident sequences, ETs and specific consequences (end states with different core degradation and/or direct releases).

The importance and uncertainty analysis have been performed as well as several sensitivity cases have been investigated. The later allow the assessment of the impact of different assumption taken in the study and evaluation of the effectiveness of the proposed modifications to the RTP features.

6. PRELIMINARY RESULTS AND INSIGHTS FROM THE PSA STUDY

6.1. Preliminary Insight for RTP

The preliminary results of the PSA for the Malaysian TRIGA research reactor indicate very low frequency of the sequences with severe core degradation. However, the frequency of sequences with minor core degradation is in the range of 1.0E-4/year. From the sensitivity analysis, it is shown that human errors are one of the dominant contributor to both accidents with major and minor core degradation.

The frequencies of both types of core degradation could be significantly reduced if formal detailed emergency operating procedures for design basis and beyond design basis conditions would become available for operators. Further insights from the PSA study are being analysed.

6.2. Some Insights Regarding Specific Features of PSA for TRIGA-type Research Reactors

One of the main insights of the study is that the PSA techniques recommended in the IAEA Safety Guide on Level 1 PSA [2] are fully applicable to the Malaysian TRIGA research reactor, which is a low power research reactor. For such reactors, however, there are specific features of their design and operation, which require consideration of specific aspects usually not applied in PSAs for NPPs. These aspects are the following:

1) A thorough analysis and detailed model of reactors scram system for both automatic and manual scram is needed. This is due to the fact that (a) in case of successful and timely scram of the reactor, the core degradation could be avoided for all IEs, and (b) automatic scram signals for many IEs (e.g. LLOCA) are absent (which is not the case for NPPs).

2) A more thorough analysis of Type B human errors leading to IEs is needed. These errors can be committed during experiments, which are part of normal activities during research reactor operation.

3) There may be a need to perform specific thermal hydraulic analyses for different IEs to confirm success criteria for end states if references to, or results of such analyses, are not addressed in the safety analysis report or in a way that it could verify the end states in the PSA study.

4) IEs frequency assessment requires extensive search for the applicable generic data, which are not publicly available.

5) The scope of the PSA study is directly depending on the design of the confinement and the nature of the working hours for the operation of the TRIGA reactors.

7. TRANSFERRING PSA SKILLS TO THE MALAYSIAN PSA TEAM

To the date, most of PSA studies for nuclear installations around the world have been completed; therefore there are some limitations to share a PSA study of any particular nuclear installation and obviously it is very difficult for new user to participate in the actual development of a PSA for a nuclear installation. The COMPASS-M project appeared to be a kind of compensation for the missing opportunities to participate in an actual PSA study.

The COMPASS-M project is aimed to nurture and exquisite the practical applications of safety assessment knowledge focusing on the development of practical skills for relevant safety assessment techniques. By implementing this project, the participants have got an opportunity to conduct the entire PSA study and address in detail all specific tasks shown in Figure 1.

In this project, a capable PSA team was formed to perform all PSA tasks. The results of the work were periodically checked by the IAEA technical experts during TRMs. This allowed providing timely feedback and correcting mistakes. Findings and comments from the reviews served as a basis for further work of the PSA; these were promptly addressed.

Because this PSA is performed for the research reactor, it might not be as complex as a PSA for an NPP, but the PSA tasks being done in this project are covering the same tasks as those ones required for a PSA of an NPP. With that understanding in mind and through the process of 'learning by doing' as demonstrated in COMPASS-M project, the transfer of the skills required to conduct a Level 1 PSA for a nuclear installation, including NPPs, has been achieved.

8. CONCLUSIONS

This paper provides information on the development process and preliminary insights from the IAEA-led project on PSA for the TRIGA research reactor in Malaysia. The project is aimed at building competence and capacity for PSA for NPPs by applying PSA techniques recommended in the IAEA Safety Guide on Level 1 PSA [2]. The PSA model was developed by the Malaysian PSA team using professional PSA software used in PSAs for NPPs. PSA results and insights will be used to further enhance the safety of the RTP and provide feedback to IAEA guidance publications on PSA for research reactors.

It is planned to extend this study to cover internal fire and floods, external hazards, and Level-2 PSA.

Acknowledgement

The authors would like to express their appreciation to the Norwegian Government for providing financial support to the PSA study. The authors would also like to extend their gratefulness to the members of the Steering Committee, whose help and advice were important for the project success: Mr. Pete Wells (IAEA, SAS, EBP Manager), Mr. David Sears (IAEA, RRSS), Mr. Mohd Pauzi Mohd Sobari (AELB, Division Manager, Malaysia), Mr. Mohamad Puad Hj Abu (MNA, Division Manager, Malaysia), and Mr. Faizal Mohamed (UKM, Lecturer, Malaysia).

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