Development of Level 1 Probabilistic Safety Assessment for Jordan Research and Training Reactor (JRTR) in the Framework of IAEA-led Competence Building Project

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In 2014, the IAEA initiated an extra budgetary competence building project COMPASS-J, aimed to support the development of technical capabilities in Jordan that is needed for the future nuclear power plant project in the area of Probabilistic Safety Assessment (PSA) by developing a PSA model for the Jordan Research and Training Reactor (JRTR), which is being commissioned currently and will reach full power operation in the near future. This project is supported by the IAEA, with funding by the European Union.

COMPASS-J represents learning by doing activity comprising periodical meetings between the Jordanian PSA team and IAEA experts to receive feedback for the intermediate models and analyses, as well as further guidance for PSA development, and homework in between the meetings. Two main outcomes were anticipated: (1) advanced competence of several stakeholders including the utility and Jordan Atomic Energy Commission (JAEC) staff to promote the appropriate use of PSA methods and its results in decision making and accident management that are applicable for the future NPP project, and (2) a PSA model and report for JRTR after the beginning of its operation, which can be used for various utility and regulatory purposes.

In this paper, an overview of the development of the PSA model for JRTR is reported highlighting the outcomes of this work. The project was divided into several tasks; i.e. initiating event analysis, accident sequence analysis, dependent failure analysis, system analysis, data analysis, human reliability analysis and other tasks. Each task was assigned to a group of participants having a task leader whose responsibility was to lead the task and report the work carried out to the IAEA experts. Several meetings were arranged with the IAEA experts who provided support and guidance to the participants during the different stages of the project. The existing JRTR-specific and generic information was integrated depending on its availability, which included information about the JRTR design, operational practices and procedures, human behavior, component reliability, etc.

The project was helpful in developing the participants' capabilities and qualifications in PSA, which can be further used for a successful nuclear power program, supporting safe operation of JRTR, and in other future nuclear projects in Jordan.

Keywords: Competence building, JRTR, JAEC, PSA, research reactor, COMPASS-J, IAEA.

I. INTRODUCTION

Jordan Research and Training Reactor (JRTR) is the first nuclear research reactor in Jordan, it has reached its first criticality on 25-April-2016 and is expected to reach full power operation in the upcoming few months. In addition to its scientific value and utilization applications such as radioisotope production and neutron activation analysis, Jordan wanted to have JRTR as the first step toward the establishment of a nuclear power program, where it will open the way to build the experience and knowledge in different fields related to nuclear energy, e.g. nuclear industry, regulations, international cooperation, etc. The safety of nuclear installations is one of the most important subjects that Jordan needs to build competence in, starting from this point an extra budgetary competence building project COMPASS-J was initiated by IAEA to develop the technical capabilities in Jordan for safety analysis of nuclear installations using the Probabilistic Safety Assessment (PSA) method.

The name “COMPASS-J” is a short for “Competence for Probabilistic Assessment of Safety in Jordan”; it was conducted on the basis of learning by doing technique by developing a realistic PSA model for JRTR. The objective of the project is to build an advanced competence of several stakeholders including the utility and Jordan Atomic Energy Commission (JAEC) staff to promote the appropriate use of PSA methods and its results in decision making and accident management that are applicable for the future NPP project, and to have a PSA model and report for JRTR after the beginning of its operation, which can be used for different utility and regulatory purposes, such as the assessment of the safety of the design and the potential for core damage, the contribution of a certain component failure to the overall risk and the importance of its periodical testing intervals.

II. SCOPE

The scope of this study is Level 1 PSA in which the design and operation of the reactor are analyzed in order to identify the sequences of events that can lead to core damage and consequently the core damage frequency is estimated, so it is focused on the reactor core and does not cover other sources of radioactive materials. The study covers internal initiating events, i.e. events that are caused by random component failures and human errors, where external initiating events such as earthquakes and floods are not included. The full power operation as is considered to be the most risk significant mode compared to other operation and shutdown modes has been considered in this study (Ref. 1).

III. SOFTWARE

The software that was used for the development of the JRTR PSA model is SAPHIRE 8, which was developed by U.S. Nuclear Regulatory Commission for the purpose of performing Probabilistic Risk Assessments (PRAs). SAPHIRE 8 stands for “System Analysis Programs for Hands-on Integrated Reliability Evaluations”; it is capable to model nuclear power plant’s response to initiating events, quantify associated core damage frequencies and identify important contributors to core damage by enabling the user to build event trees and fault trees, to obtain minimal cut sets and to quantify the model (Ref. 2). This software fully covers the needs of a research reactor PSA, and the respective authorization for its use was received by the Jordanian participants of the project.

IV. BRIEF DISCRIPTION OF JRTR AND ITS SAFETY SYSTEMS

JRTR is a multipurpose open pool in tank type research reactor with a nominal power of 5 MW, it has 18 fuel assemblies each with 21 plate type fuel plates. The fuel is low Enriched Uranium (LEU) with a ^{235}U enrichment of 19.75 weight %. Each fuel plate is composed of a fuel meat with surrounding aluminum cladding. The fuel meat is made of fine and homogeneous dispersion of U_3Si_2 particles in a continuous Aluminum matrix with a Uranium density of 4.8 g U/cm3. For the initial core the fuel is with 4 different Uranium densities (Ref. 3).

There are two kinds of reflectors in the JRTR. The beryllium reflector assemblies are located in the core region as the primary reflector material. The heavy water, as the secondary reflector material, is contained in the heavy water vessel. The Beryllium Reflector Assemblies (BRAs) are supported by and located on the grid plate.

There are four Beam Port Assemblies (BPAs) in JRTR, in addition to many vertical irradiation facilities submerged in the reactor pool and a dry irradiation facility of Thermal Column (TC). The majority of the vertical irradiation facilities are irradiation holes in the Be blocks and in the heavy water tank. Additional three holes are provided in the TC extension. JRTR also does have a three Neutron Activation Analysis positions with a three pneumatic transfer systems that rapidly transfers samples between them and the neutron activation analysis facility (Ref. 3).

There are two kinds of reactivity control mechanisms in JRTR; Control Rod Drive Mechanism (CRDM) and Second Shutdown Drive Mechanism (SSDM). A CRDM inserts, withdraws, or maintains at required position Control Absorber Rods (CARs) using a stepping motor, JRTR has four Hafnium (Hf) CARs where three of them can successfully shutdown the reactor. SSDM as an alternate and independent shutdown mechanism, providing a secondary means of reactor shutdown by the gravity drop of Second Shutdown Rods (SSRs). There are two B_4C SSRs in JRTR, both of them are needed to shutdown the reactor successfully if all of the CARs are assumed to be stuck. All CARs and SSRs are dropped by gravity when a reactor trip is required by the Reactor Protection System (RPS) or by the Alternate Protection System (APS) (Ref. 3).

For the cooling of the reactor core during normal power operation mode, the Primary Cooling System (PCS) is used to circulate the primary cooling water downward through the core; the heat is then transferred to the secondary cooling system through heat exchangers to be then rejected to the environment using a cooling tower. After the shutdown of the reactor, decay heat is removed by the establishment of natural circulation through two flap valves that are opened passively after stopping the PCS pumps, the flap valves are installed at the PCS outlet pipe inside the reactor pool. JRTR has two siphon break lines each with two siphon break valves at PCS inlet and outlet pipes, these siphon break valves are used to stop the leakage of the primary coolant in the PCS piping system if happened by stopping the siphon effect that results from the elevation difference between the pool and PCS piping system. Siphon break valves (the two on the pool outlet pipe) can be used to establish a natural circulation flow path if the flap valves fail to open (Ref. 3).

In case of large loss of coolant from the pool water due to multiple beam tube rupture; the Emergency Water Supply System (EWSS) is used to inject water into the reactor core in order to maintain the minimum pool water inventory that is required to prevent core uncovering and to remove the decay heat after the reactor shutdown. There are some other connected systems that have non-safety functions, e.g. Pool Water Management System (PWMS), Hot Water Layer System (HWLS), and Heavy Water System (HWS) (Ref. 3).

The electrical system of JRTR works as a support system for the proper functioning of the front line safety systems that directly perform safety function in addition to its other non-safety functions. It provides the electrical power needed to operate the different components of JRTR systems. The electrical system can be categorized into three subsystems i.e. normal power supply system, essential power supply system, and uninterruptible AC and DC power supply system. The normal power supply is connected to the electric power company; the essential power is connected to the normal power and has a backup power supply by a diesel generator, while the uninterruptible power supply is connected to the essential power and to uninterruptible AC system and a battery as uninterruptible DC power supply system (Ref. 3).

V. OVERVIEW OF THE JRTR DEVELOPMENT PROCESS

The development of JRTR PSA was performed following the procedures published by IAEA, such as IAEA Safety Guide SSG-3 (Ref. 1). The participants of COMPASS-J project were divided into groups; each group was assigned different PSA task. The project included the following main tasks:

1) Plant familiarization and information collection
2) Initiating event analysis.
3) Accident sequence analysis.
4) Dependent failure analysis.
5) System analysis.
6) Data analysis.
7) Common cause failure analysis.
8) Human reliability analysis.
9) Analysis and interpretation of the results (sensitivity studies, importance and uncertainty analysis).

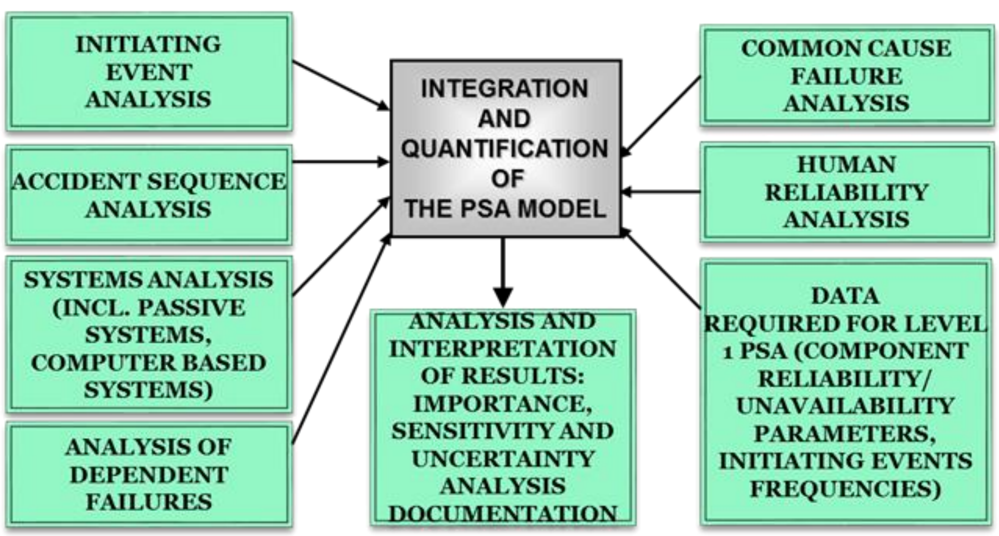

Fig 1 PSA tasks

V.A. Familiarization with the plant and information collection

In this task, the PSA team familiarized themselves with a general information about the design and operation of JRTR by collecting the needed documents and specific information depending on the mentioned scope of this PSA study as a Level 1 internal events PSA, this included studying JRTR Safety Analysis Report (SAR) (Ref. 3), Operational Technical Specifications (OTS) (Ref. 4), system description and design specifications, in addition to operation procedures of JRTR including normal, abnormal and emergency procedures. Also the PSA team was familiarized with the electrical single line diagrams and system drawings of fluidic and I&C safety systems such as RPS and APS to understand their circuits and components connections and dependency in details. Information about the success criteria of different safety systems and components was collected. Several discussions were made with JRTR operation staff to get some needed information and technical specifications of JRTR.

V.B. Initiating events analysis

An initiating event is defined as "an event that could directly lead to core damage (e.g. reactor vessel rupture) or challenges normal operation and that requires successful mitigation using safety or non-safety systems to prevent core damage" (IAEA Safety Guide on Level-1 PSA, SSG3) (Ref. 1).

In order to identify the initiating events for the JRTR PSA, the recommendations of IAEA Safety Guide SSG-3 were followed, the approaches that have been applied after plant familiarization included an analytical engineering method such as Failure Modes and Effect Analyses (FMEA) to determine whether system/components failures, either partial or complete, could lead to an initiating event (IE), then a Deductive analyses such as Master Logic Diagrams (MLD) was applied to determine the elementary failures or combinations of elementary failures that would challenge normal operation and lead to an IE. After that the initiating events identified were compared to those included in generic initiating events list available in the IAEA TECDOC reports on research reactors, and to operational experience data from other research reactors. Finally the list of IEs compiled at previous stages was compared with the list of postulated IEs considered in the FSAR.

About forty initiating events were generated using the previously mentioned approaches; these initiating events were later grouped into ten groups as follows:

1) Loss of Flow Accident (LOFA)/Failure of the two PCS pumps.
2) Partial Loss of Flow Accident (PLOFA) /Failure of one PCS pump.
3) Loss of Offsite Electric Power (LOEP).
4) Loss of Secondary Cooling (LOSC).
5) Reactivity Insertion Accident (RIA).
6) Small Loss of Coolant Accident (SLOCA)/ loss of coolant accident in PCS (outside the pool).
7) Core bypass (BYPASS)/ due to PCS pipe rupture inside the pool or spurious opening of flap valve or SBV during power operation.
8) Large Loss of Coolant Accident (LLOCA) / due multiple beam tube ruptures.
9) General transients (GTRN).
10) Flow blockage (BLOCK).

V.C. Accident sequence analysis

In the accident sequence analysis; the purpose is to identify the accident progressions following IEs selected for further modelling at the stage of IEs analyses, and to determine the consequence of each accident sequence.

As this study is a Level 1 PSA which is subjected to the assessment of the reactor facility failures leading to the core damage, the consequences considered here are only the core damage consequences. What can lead to the core damage in the case of JRTR is mainly the failure to satisfy one of the safety functions of reactivity control, core cooling and maintaining the inventory of the cooling system.

In this study the core damage states are classified into four states; first Core Damage state (CD1), second Core Damage state (CD2), third Core Damage state (CD3) and fourth Core Damage state (CD4). CD1 was defined as the state that results when transient happens without reactor scram or in the case of reactor core uncovering due to large loss of coolant accident. In this state, all the fuel assemblies in the reactor will be damaged. In CD2, it is assumed that a transient occurs with scram and no means of decay heat removal available. In this state, the fuel in the reactor may be partially damaged due to lack of primary cooling or natural circulation cooling. CD3 was defined as the state in which unstoppable loss of coolant accident happens where the reactor is successfully scrammed and the EWSS starts to provide cooling by injecting water into the core, in this case minor degradation is assumed as a conservative approach till a thermal hydraulic analysis is made to ensure that natural circulation of the water injected by EWSS in the Reactor Structural Assembly (RSA) will be sufficient to remove the residual decay heat of the core. CD4 was defined as fuel degradation due to blockage of one fuel assembly; here the core damage can only happen in the blocked fuel assembly. Event trees were built in order to identify the accident sequences that would lead to the first, second, third and fourth core damage states.

In case of the occurrence of any of the initiating events, JRTR safety systems shall take an action to reliably satisfy the safety functions, these safety systems are:

1) Reactor Protection System (RPS).
2) Alternate Protection System (APS).
3) Primary Cooling System (PCS).
4) Reactor pool (for natural circulation).

5) Emergency Water Supply System (EWSS).
6) Siphon Break Valves (SBV), for pool isolation.

Small event tree and large fault tree approach was used in modeling the accident sequences. The accident sequence model was developed for all of the ten initiating events with fault trees top events to be failure to satisfy one of the mentioned safety systems functions such as failure to scram the reactor, failure of primary cooling, failure of natural circulation or failure of pool isolation.

In JRTR there are two modes of operation; the training and power operation modes. In this study the focus was on studying full power operation mode as in the training mode the reactor is much more simply operated and safer than the power operation as the power is low. Table I bellow shows a simple comparison between the system response to the initiating events in the power and training modes of operation.

Table I System response to initiating events in the power and training modes of operation

Initiating event	System response in power operation mode	System response in training mode
Loss of Flow Accident (LOFA)	RPS/APS will trip the reactor, and the core natural circulation will start through the FVs or SBVs.	Not applicable (During the training operation the reactor is cooled by NC).
Partial Loss of Flow Accident (PLOFA)	RPS/APS will trip the reactor, and the core natural circulation will start through the FVs or SBVs.	Not applicable (During the training operation the reactor is cooled by NC)
Loss of Offsite Electric Power(LOEP)	The reactor will be tripped passively, and the core natural circulation will start through the FVs or SBVs.	The reactor will be tripped passively. FVs and SBVs are already opened for natural circulation.
Reactivity Insertion Accident (RIA)	RPS/APS will trip the reactor, and the core natural circulation will start through the FVs or SBVs.	RPS/APS will trip the reactor. FVs and SBVs are already opened for natural circulation.
Small Loss of Coolant Accident (SLOCA)	RPS/APS will trip the CARS, while SSRs will be tripped passively, the SBVs open to stop leakage, and the core natural circulation will start through the FVs.	RPS/APS will trip the CARs, SSRs will be tripped passively, the SBVs open to stop leakage, and FVs are already opened for natural circulation.
Core bypass (BYPASS)	RPS/APS will trip the reactor, and the core natural circulation will start through the FVs or SBVs.	Not applicable (During training operation pumps are off).
Large LOCA (LLOCA)	RPS/APS will trip the reactor and the EWSS will start to inject water to the core.	RPS/APS will trip the reactor and the emergency water supply system will start to inject water to the (less severe consequences).
Loss of Secondary Cooling (LOSC)	RPS/APS will trip the reactor, and the core natural circulation will start through the FVs or SBVs.	Not applicable (Secondary cooling system is not needed during training mode)
Flow blockage	RPS/APS will trip the reactor after detection of core damage, and the core natural circulation will start through the FVs or SBVs	The power is low, so the core damage is not expected in this situation.

V.D. System analysis task and dependent failure analysis

In order to prevent the reactor core damage that can happen following the occurrence of an initiating event, the fundamental safety functions should be maintained. These functions include control of reactivity, cooling of the core and maintaining the inventory of the cooling system.

These safety functions are incorporated in JRTR using several safety systems. For the function of control of reactivity, the safety systems that are used are the RPS and APS; that are responsible for the reactor shutdown by dropping CARs and SSRs in case of the occurrence of an initiating event.

For the core cooling, PCS is used in the power operational mode, while the natural convection of the reactor pool water through the flap valves is used in the case of reactor shutdown or training operational mode.

One of the most important issues requiring consideration is to maintain the inventory of the cooling system. This can be accomplished by using the siphon break valves system which are actuated by the RPS for the isolation of the pool water from leakage, and the EWSS which is actuated to provide make-up water to cover the reactor core with water for a sufficient period of time when a loss of pool water accident occurs by multiple beam tube simultaneous rupture.

The systems interdependencies were identified and documented in the form of matrices. In the reactor we have two kinds of systems to serve each of the safety functions; the systems that directly perform a safety function are termed front-line systems; while those required for the proper function of the front line systems are termed support systems.

In JRTR the front line systems are: RPS, APS, PCS, reactor pool, EWSS and SBVs. Table II below shows the front line systems and the related safety functions.

Table II Safety function and corresponding front-line system

Safety function	Front line system
Control of reactivity	Reactor protection system (RPS) Alternate protection system (APS)
Heat removal	Primary cooling system (PCS) Reactor pool Emergency water supply system (EWSS)
Maintaining pool inventory	Siphon break valves (SBV) Emergency water supply system (EWSS)

The system dependency matrices of the front line on front line systems, front line on support systems and support on support systems were developed and analyzed.

After the comprehensive study of the system a detailed fault tree was built for each safety function. The top events of the developed fault trees are the failures related to the mentioned safety functions, the top events included failure to scram the reactor, failure of primary cooling, failure of natural circulation, failure of pool isolation by siphon break valves and failure of the EWSS to inject water into the core in the case of LLOCA.

For a specific safety functions, several fault trees were developed depending on the system response for the initiating event, for example in the case of the reactor scram due to LOEP initiating event, the reactor will be tripped passively by CARs and SSRs without any need to the RPS or APS, as the electrical magnet of CRDM and the pumps of SSDM will release the control rods passively by losing the power. The same situation is in the case of LOCA where the SSDM (but not CRDM) will shutdown the reactor passively as the water goes bellow the suction of the SSDM pumps without the need for RPS or APS actuation signal. Another example is the natural circulation in the case of SLOCA were siphon break valves are not useful here for natural circulation as the pool water level will be lower than siphon break line level in the reactor pool not allowing natural circulation to be established through them. House events were used for the modeling of components that are only needed for specific initiating events such as the RPS and APS transmitters.

The fault trees were developed in a detailed manner, where every component failure in the system was considered, taking RPS as an example; all of its control and actuation circuit relays, transmitters, push buttons, etc. were considered in the model. Human failure to take a recovery action was considered as well, e.g. failure of the manual SCRAM or manual actuation of SBVs.

As a conservative approach, operator success to SCRAM the reactor or to actuate SBVs was linked to the success of the transmitters to provide a signal to Main Control Room (MCR), i.e. it was assumed that the operator cannot shutdown the reactor manually unless he gets some signal delivered to him by the transmitters.

V.E. Data analysis

As there is no operational history for the facility at this time, no specific data for the JRTR have been used, and the whole analysis was basically based on the generic data available in the recognized references and using justified engineering judgments.

V.E.1. Initiating events frequency

In the course of development of IE frequencies, only internal IEs have been considered. Such IEs include events caused by systems and equipment random failures or personnel erroneous handling. Considered IEs also include loss of external supply of electrical power to the facility.

For each of the IEs the following sources and methods have been used to obtain a representative IE frequency:

1) Generic operational statistical data of operational abnormalities.
2) Reliability models, for IEs whose occurrence is caused by systems' equipment failure during the operational mode analyzed the frequency is calculated using simplified logical models with component reliability data.
3) Human reliability models; for IEs whose occurrence is caused by human unintentional errors which are considered as root causes for IEs, frequencies were calculated using human reliability models.

IEs' frequencies used in the analysis are provided in the Table III with their description below.

Table III IEs' frequencies used in the analysis.

#	ID	Description of the IE	Frequency / [year]
1.	**LOFA**	Loss Of Flow Accident/Failure of two PCS pumps	1.98E-02
2.	**PLOFA**	Partial Loss Of Flow Accident/Failure of one PCS pump	1.29E-01
3.	**BLOCK**	Flow blockage in one fuel assembly	1.00E-02
4.	**LOEP**	Loss of Offsite Electrical Power	5.00E-01
5.	**LOSC**	Loss Of Secondary Cooling	1.98E-02
6.	**RIA**	Reactivity Insertion Accident	4.20E+00
7.	**SLOCA**	Small Loss Of Coolant Accident	5.00E-04
8.	**BYPASS**	Coolant core Bypass accident / PCS pipe rupture inside the pool or spurious opening of siphon-break valves during power operation	1.30E-03
9.	**LLOCA**	Large Loss Of Coolant Accident/Multiple beam tubes rapture	5.00E-06
10.	**GTRN**	Erroneous handling	1.02E-02

V.E.2. Components' Failure Rates

In the course of development of components' reliability database the modeled front-line and support systems were identified. All the components which were modeled in the fault trees have been listed and grouped according to the similarity in their:

1) Type and design.
2) Function.
3) Failure modes.

As the facility was not operational at the time of the analysis, no JRTR-specific operational data have been used and the reliability data for components were calculated using generic data mainly from proper references Refs. 5, 6, 7. The data retrieved from the references was in the form of:

1) The number of components of the same type considered in the database.
2) The number of failures during the exposure time.
3) Components' exposure time used in the reporting of the data.

If statistical data were not available in the literature Refs. 5, 6 and 7 for the considered equipment, and the failure rate was reported, the failure rate was retrieved as it is. If the failure rate was not available, engineering judgment considering applicable information was applied.

Similar components were grouped together and same data was used for one group members. The data collected for each group of components included:

1) Number of components for which the data was collected in the reference.
2) Cumulative calendar time.
3) Cumulative operating time.
4) Total number of demands.
5) Total number of failures.

To calculate the failure rate per demand and the failure rate per hour the following equations were used:

$$prob/demand = \frac{Total\#of\_failures}{Total\#of\_demands}. \qquad (1)$$

$$\lambda/hour = \frac{Total\#of\_failures}{Adjusted\_Exposure\_Time}. \qquad (2)$$

Where:
Adjusted Exposure Time= Cumulative operating time + δ x (Cumulative calendar time)

λ: the failure rate
δ: a modification factor which took two values: 0.3 and 1. This factor modifies the time to consider the operating time within the cumulative time if only total installation time was provided in the reference.
An *F distribution* was assumed to calculate the confidence intervals for prob./demand using the following equations:
For 5%:

$$\frac{\#Failures \times (Inverse\_F(1-0.95, 2\times\#Failures, 2\times\#Demands - 2\times\#Failures + 2)}{(\#Demands - \#Failures + 1) + \#Failures \times (inverse\_F(1-0.95\times\#Failures, 2\times\#Demands - 2\times\#Failures + 2)}. \qquad (3)$$

For 95%:

$$\frac{\#Failures \times (inverse\_F(1-0.05, 2\times\#Failures, 2\times\#Demands - 2\times\#Failures + 2)}{(\#Demands - \#Failures + 1) + \#Failures \times (inverse\_F(1-0.95, 2\times\#Failures, 2\times\#Demands - 2\times\#Failures + 2)}. \qquad (4)$$

And a *Chi-square distribution* was assumed to calculate the confidence intervals for λ/hour using the following equations:
For 5%:

$$Inverse\_ChiSquare(1-0.05, \frac{2\times\#Failures + 2}{2\times Cumulative\_time}). \qquad (5)$$

For 95%:

$$Inverse\_ChiSquare(1-0.95, \frac{2\times\#Failures + 2}{2\times Cumulative\_time}). \qquad (6)$$

The model was constructed considering about **300** different components, of these only **8.5%** were considered as mechanical parts, **20.5%** as instrumentation and control parts, and **71.5%** as electrical or electro-mechanical parts.

V.E.3. Common Cause Failures

In order to incorporate the impact of Common Cause Failure (CCF) into systems' reliability models, component groups were identified by the analyst based on the following attributes:

1) Component type.
2) Component use and function.
3) Component failure modes.
4) Housing systems' interfaces.
5) Component environmental conditions.
6) Component maintenance characteristics.

Whenever an appropriate combination of components was noticed which shares one or more of the above features, the combination was analyzed for a potential CCF. A maximum number of six members inside one group were considered. This number originated from the design-specifications of the modeled system; e.g. each of the Coincidence, Initiation and Actuation circuits is composed of six relays which could malfunction due to a CCF.

For the purpose of handling different levels in the redundancy, especially in shutdown systems (RPS and APS), Alpha Factor model was considered to model the CCFs.

Depending on the number of members inside each group and the type of components, the most suitable alpha factors set was retrieved from U.S. Nuclear Regulatory Commission, "CCF Parameter Estimations, (Ref. 8). If not applicable, general alpha factors were retrieved from Idaho National Engineering and Environmental Laboratory document: "Common-Cause Failure Analysis for Reactor Protection System Reliability Studies" (Ref. 9), which depends only on the number of components inside a group.

V.F. Human Reliability Analysis

An expanded study of the Human Reliability Analysis (HRA) discussing the three human actions types according to the classification of IAEA safety series No. 50-P-10 (Ref. 10); Types A and B (including the operation and training modes), and concentrating on the Type C human actions to study the possible human actions after the following initiating events: LOEP, RIA, LOFA, LOCA, fuel channel blockage and GTRN.

The HRA done for the JRTR contains a detailed analysis for all probable human actions of Types A, B and C of the human actions classification. The quantification process used to calculate Types A and B human errors probabilities is based on the Technique for Human Error-Rate Prediction (THERP). For Type A human errors, the probabilities were calculated if these errors will lead to unavailability of any system or component during operation. For Type B human errors that could lead to an initiating event, the probabilities were calculated for all of them. For some Type B human errors, the recovery mechanism is assumed, because there is a possibility of a recovery of the error, as checks by a second person, receipt of new indications, post-maintenance tests, and arrival of new personnel. The human actions Type C following the accidents are analyzed to evaluate the probabilities of the human failures. The analysis process is done based on one of the HRA techniques; the Standardized Plant Analysis Risk (SPAR) HRA (SPAR-H) Method based on NUREG/CR-6883 (Ref. 11). The analysis have been done and the probabilities calculated for human actions Type C for the postulated Control Room Scenario (Operator Actions) after the initiating events LOEP, RIA, P-LOFA, P-LOFA, LOSC, S-LOCA, Core Bypass, L-LOCA, and GTRN.

There are many interlocks in the design of the JRTR helping to reduce the severity of human failure actions, some of them are enough to reduce the consequences of the human error from Type B to Type A. Even though that the Type A human errors do not cause an initiating event, but it is important to study them because some errors could lead to unavailability of some important systems during any accident which make the accident progression more severe.

The resulted Type B human failures from the JRTR human actions analysis are few and the main sequence is the RIA, some human failures may cause LOSC, some may cause LOFA, and some may cause GTRN. They are few because the design is based on the accidents that occurred in reactors around the world due to human failures, and the human factor has been considered in the JRTR design.

Type C human failures probabilities were analyzed including human diagnoses and actions. In the JRTR, the diagnoses part is the dominant in the final probability because the human actions are simple and the operator know well how to manually do these actions, but he needs more time and work on the diagnose after the accidents.

VI. PRELIMINARY RESULTS

The accident sequences leading to core damage were quantified in order to evaluate the core damage frequency and dominant contributors to core damage. The core damage frequency of each of the four end states which represents the summation of the frequencies of all the event tree sequences leading to that state was estimated. For the analysis of the fault tree and accident sequence a truncation value of 1.00E-15 was used.

The point estimate values for the core damage frequency (CDF) were 3.85E-06 for CD1, 4.43E-07 for CD2, 5.00E-06 for CD3 and 1.00E-02 for the CD4.

Fig 2 and Fig 3 bellow show the contribution of the different initiating events in the core damage states CD1, and CD2. The only contributor in CD3 and CD4 are LLOCA and BLOCK initiating events respectively. It can be noticed that the major contributor to CD1 is SLOCA, while the major contributor for CD2 is PLOFA, as CD2 was taken as the end state for PLOFA in both cases of failure to SCRAM and failure to provide decay heat removal.

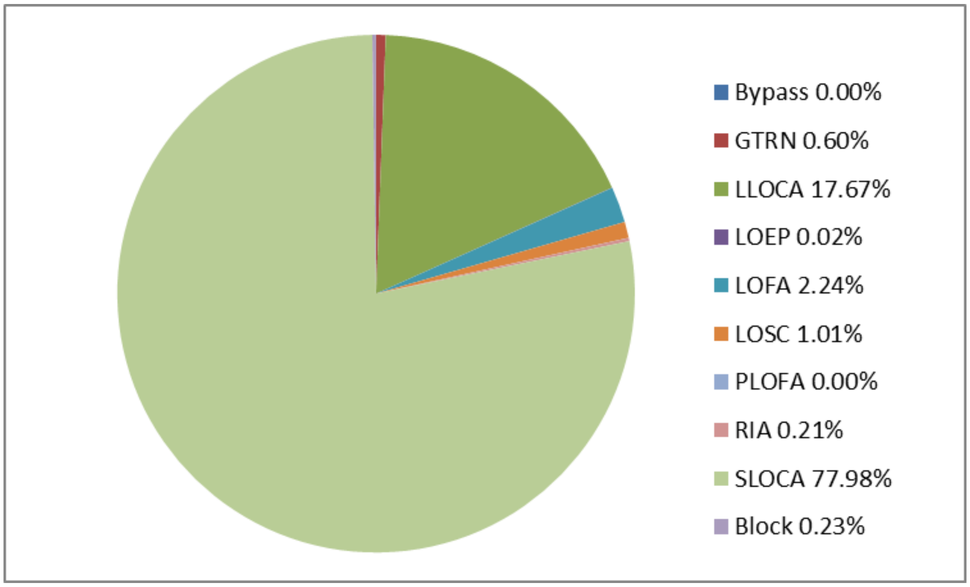

Fig 2 Contribution of the different initiating events in the core damage frequency (CD1)

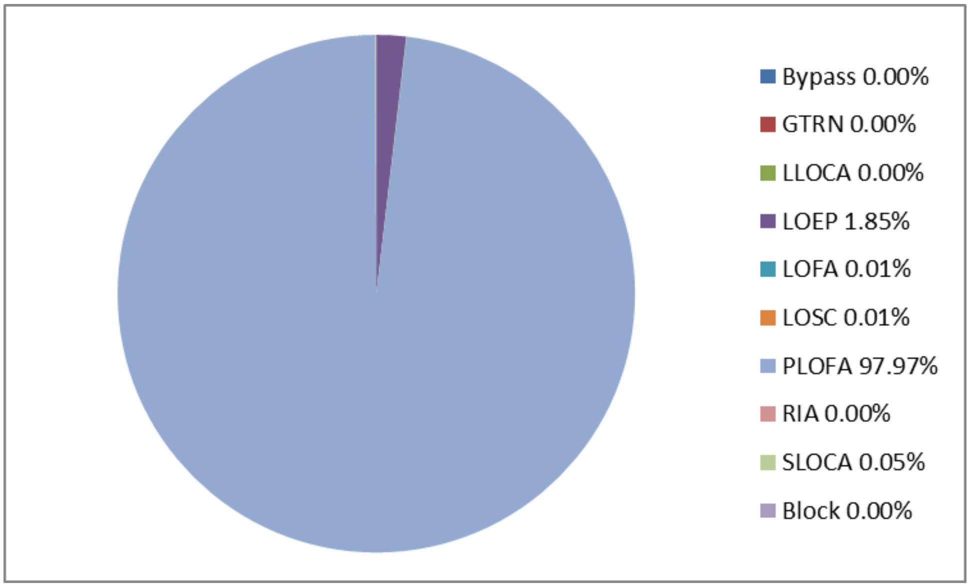

Fig 3 Contribution of the different initiating events in the core damage frequency (CD2)

According to the importance analysis of CD1 it was found that the most important event to happen is the common cause failure of the four siphon break valves, while the most important event for CD2 is the recovery failure to trip the reactor by the operator using the APS.

VII. CONCLUSIONS AND FUTURE WORK

The PSA study for JRTR is the first domestic PSA study for a nuclear facility in Jordan. It was done in the framework of IAEA-led competence building project to train the Jordanian specialists for advanced safety assessment in the view of the preparation for nuclear power plant project.

From the preliminary results, it was noticed that the partial core damage frequency that may result from the fuel blockage of one fuel assembly (CD4) is not negligible and should be taken into account more seriously. A further study and investigation should be done on how likely this event is, and to evaluate possible measures if found appropriate.

As a conservative approach it was assumed in this study that even in the case of success of EWSS to inject water in LLOCA, CD3 will happen. Actually this will depend on the leakage rate of coolant out of the RSA; further thermal hydraulic analysis shall be done to determine the possibility of core damage in this situation with different leakage rates from the RSA.

The calculation has shown a high importance of the transmitters' periodical test. The transmitters' periodical test is every 45 days, but the calibration is every one year. A further investigation should be done to evaluate risk significance of calibration on the functionality of the transmitters.

Further research will be beneficial for evaluation of the frequency of RIA. More detailed analysis of PSA results and major contributors to the risk of core damage may provide support for further safety substantiation.

The PSA research will be continued to cover other hazards (e.g. seismic) and safety in training and shutdown modes. The further work includes also the implementation of minor adjustments to the PSA report based on the recommendations of the technical review meeting with IAEA experts and to finalize the PSA report.

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