Journal of the Korean Society of Safety
Vol. 34, No. 3, pp. 102-110, June 2019
https://doi.org/10.14346/JKOSOS.2019.34.3.102
ISSN 1738-3803 (Print)
ISSN 2383-9953 (Online)
http://www.kosos.or.kr/jkosos

# 연구용원자로 기본설계에 대한 예비 확률론적 안전성 평가

이윤환†

# Aspects of Preliminary Probabilistic Safety Assessment for a Research Reactor in the Conceptual Design Phase

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**Received** : April 9, 2018
**Revised** : June 22, 2018
**Accepted** : February 15, 2019



**Abstract** : This paper describes the work and results of the preliminary Probabilistic Safety Assessment (PSA) for a research reactor in the design phase. This preliminary PSA was undertaken to assess the level of safety for the design of a research reactor and to evaluate whether it is probabilistically safe to operate and reliable to use. The scope of the PSA described here is a Level 1 PSA which addresses the risks associated with core damage. After reviewing the documents and its conceptual design, eight typical initiating events are selected regarding internal events during the normal operation of the reactor. Simple fault tree models for the PSA are developed instead of the detailed model at this conceptual design stage. A total of 32 core damage accident sequences for an internal event analysis were identified and quantified using the AIMS-PSA. LOCA-I has a dominant contribution to the total CDF by a single initiating event. The CDF from the internal events of a research reactor is estimated to be 7.38E-07/year. The CDF for the representative initiating events is less than 1.0E-6/year even though conservative assumptions are used in reliability data. The conceptual design of the research reactor is designed to be sufficiently safe from the viewpoint of safety.

**Key Words** : PSA, research reactor, core damage frequency, CDF

## 1. Introduction

This paper describes the work and results of the preliminary Probabilistic Safety Assessment (PSA) of a research reactor in the design phase. The PSA has been performed to assess the level of safety for the design of a research reactor and to evaluate whether it is probabilistically safe to operate and reliable to use according to the procedures published by IAEA[1,2] and US NRC[3].

### 1.1. Objectives of this study

This preliminary PSA was undertaken to assess the level of safety for the design of a research reactor and to evaluate whether it is probabilistically safe to operate and reliable to use. The technical objective of this study was to identify accident sequences that lead to core damage and the corresponding frequencies.

The PSA has the following primary objectives:

(1) To provide a mechanism for assuring a balanced design from a risk perspective (i.e., such that there are no outliers or individual features that contribute a large fraction of the overall risk).
(2) To identify the leading core damage and risk sequences.
(3) To identify potential vulnerabilities to core damage for a reactor design.
(4) To identify the risk-important safety-related systems or components to support the definition of meaningful technical specifications.
(5) To serve as the basis for an accident management program

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### 1.2 A Research Reactor

A research reactor is a multi-purpose open-tank-in-pool type reactor as shown in Fig. 1. The reactor and its major components are submerged in the reactor pool. The core is cooled by a flow of demineralized light water in the reactor pool flowing downward direction in forced-circulation cooling mode. When the reactor is shutdown, the decay heat of the core can be removed by natural convection. For this purpose, flap valves are installed in the outlet pipe of the primary cooling system and the primary coolant is hydraulically connected with the pool water when the primary cooling pumps stop.

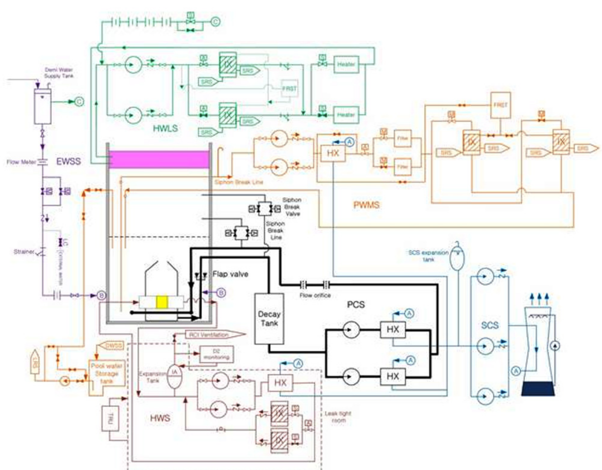


Fig. 1. Schematic diagram of reactor cooling and connected system of a research reactor.

### 1.3. Scope of PSA

The PSA has the following scope:

(1) The scope of the PSA reported here is a Level 1 PSA which addresses the risks associated with core damage. It includes an evaluation of the types of accidents that could lead to core damage, and an assessment of their frequencies.

(2) The PSA includes only internal initiating events including a loss of offsite power. External initiating events like earthquakes, floods, fires, and sabotage are not included in this study.

(3) All operating modes of the reactor have been considered but only a full-power operation has been assessed as risk significant.

(4) Core damage has been conservatively assumed to result in any state of the core where the fuel temperature exceeds the design limit, or if the available thermal-hydraulic analyses cannot demonstrate a successful cooling of the core.

(5) The PSA employs traditional PSA techniques for a quantitative evaluation of plant risks and uses generic data, particularly those reported in IAEA publications[4].

This paper presents the methodology and software for PSA (Section 2), the identification of accident initiators (Section 3), event tree analysis (Section 4), fault tree analysis (Section 5), data analysis (Section 6), accident sequence quantification (Section 7), and concluding remarks (Section 8).

## 2. Methodology and Software for PSA

### 2.1. Scope of PSA

The PSA has the following scope: This preliminary PSA for a reactor design uses methodologies consistent with those outlined in the 'IAEA PSA Procedures Guide' [1]and 'PRA Procedures Guide'[2] The following sections describe the major tasks of the PSA. There are six major tasks associated with the Level 1 internal event analysis, as shown in Fig. 2.

The first task in the PSA is plant familiarization. The objective of this task is to collect the information necessary for performing the PSA. All information necessary for identification of appropriate initiating events, determination of the success criteria for the systems required to prevent or mitigate the transients and accidents and to identify the dependence between the front line systems and the support systems which are required for a proper functioning of the front line systems.

The second one is to identify and select postulated initiating events. An initiating event is regarded as an event, which may lead to the core damage if it is combined with the failure of the safety features.

The third one is to develop various accident scenarios (accident sequences) that are combinations of the initiating events and the successes or failures of the systems. This task is accomplished using the event tree.

The fourth task is system modeling. The task includes the construction of fault tree models to identify the causes and probabilities of the system failure. A fault tree of a system is the logical representation in which various causes of a system failure are combined using logic gates, such as OR and AND. Component hardware failures, common

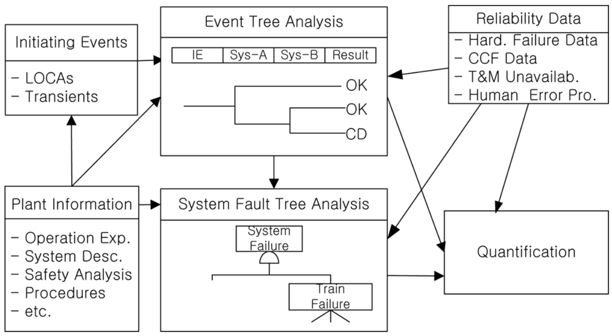


Fig. 2. Major tasks of the PSA.

cause failures, human errors, and unavailability due to testing and maintenance are included in the fault tree model for a postulated system failure.

The fifth involves collecting and evaluating the reliability data required for the quantification of event tree and fault tree models. The type of reliability data includes the initiating event frequencies, component hardware failure rates, common cause failure rates, and human error probabilities.

The final task is the accident sequence quantification. The objective of the accident sequence quantification is to evaluate the Core Damage Frequency (CDF) for each sequence and to find the dominant contributors to the risk. The event tree and fault tree linking approach is used as a basic method of the accident sequence quantification. The total CDF is estimated as the sum of the frequencies of the individual accident sequences resulting in core damage.

## 2.2. PSA Software

The PSA has the following scope: This preliminary PSA for a reactor design uses consistent methodologies. The AIMS-PSA (Release 2, version 1.0)[5] and FTREX[6] are used for the preliminary PSA of a research reactor. The AIMS-PSA developed by the Korea Atomic Energy Research Institute (KAERI) is software for a PSA. It provides a tool to construct fault trees and event trees, to generate minimal cut sets for each sequence, and to perform the importance and uncertainty analyses. The AIMS-PSA is the updated version of the KIRAP[7] which has been used for a number of PSAs performed in Korea.

The AIMS-PSA was developed to simplify the PSA works. If a PSA model is provided, the AIMS-PSA integrates the PSA model to build one fault tree model for an evaluation of the whole core damage frequency, and also generates minimal cut sets for the core damage frequency. Only few button clicks are required to perform the quantification of the PSA. This helps a PSA analyst to perform the PSA easily and quickly. The cut set generation engine FTREX[6] developed by KAERI is used for the AIMS-PSA. The FTREX is the most powerful cut set generator that has been successfully used for many PSAs or risk monitors in Korea and the USA. The accident sequence quantification for this PSA can be conducted in a few seconds. The FTREX has many useful features which are to generate minimal cut sets from a fault tree with circular logic, and to perform a rule-based recovery analysis.

# 3. Identification of Accident Initiators

## 3.1. Determination and Selection of POS

The following plant operation states (POSs) have been considered at the conceptual design stage.

(1) Nominal full power operation (5MW)
(2) Reduced power operation
(3) Start-up operation
(4) Shutdown operation

Nominal full power operation is a plant operating state bracketing all other states from the safety point of view. This is due to the fact that the reactor pool constitutes a large heat sink that is always available, regardless of the operating state of the reactor. Therefore, this PSA has been performed for the state of the nominal full power operation.

## 3.2. Identification of Initiating Events

An initiating event is an event that leads to an unplanned reactor trip and requires a mitigating action on the part of automatic systems or the operator in order to maintain the safety functions of the plant. Component failures or human errors can be an initiating event. In order to identify the initiating events for the PSA, the following approaches were applied after plant familiarization.

The first approach applied to identify initiating events is the logical evaluation which develops the Master Logic Diagram (MLD). The MDL is a high level fault tree model of the potential causes of a postulated undesirable event and the logical relationships between these potential causes. A set of conceptual core damage general initiators can be identified from the MLD. The second approach for identifying initiating events is to list up all postulated events from intensive reviews on design documents including Failure Mode and Effect Analysis (FMEA), safety analysis reports, and other reference to previous lists for research reactors.

### 3.3. Safety Functions and Corresponding Safety Systems

The design of a research reactor incorporates a number of safety functions aiming at preventing core damage to occur following an initiating event. Table 1 shows three safety functions and corresponding safety systems in order to prevent core damage.

Table 1. Safety functions and corresponding systems

| Safety functions | Corresponding systems |
|---|---|
| Control reactivity | Reactor protection system & Alternate protection system<br>a. First shutdown system<br>b. Second shutdown system |
| Maintain primary coolant inventory | Reactor pool isolation and make up<br>a. siphon Break Valves<br>b. Emergency water supply system |
| Remove core decay heat | a. Primary cooling system<br>b. Reactor pool natural convection<br>c. Emergency water supply system |

### 3.4. Grouping of Initiating Events

Based on the responses of the corresponding safety systems, the initiating events can be grouped in such a way that all events in the same group impose essentially the same success criteria on the safety systems as well as the same specific conditions.

The following initiating events have been finally selected for the development of an event tree in the preliminary PSA for a research reactor.

- Loss of electric power (LOEP)
- Reactivity insertion accident (RIA)
- Loss of primary cooling flow (LOPCS)
- Loss of secondary cooling flow (LOSCS)
- Loss of coolant accident - out pool (LOCA-I)
- Loss of coolant accident - in pool (LOCA-II)
- Beam tube LOCA (LOCA-III)
- General transients (GTRN)

Table 2 shows the initiating events finally selected for the preliminary PSA of a research reactor.

Table 2. Initiating events of preliminary PSA for a research reactor

| Initiating events | Descriptions |
|---|---|
| 1. LOEP | Rx trip due to offsite power |
| 2. RIA | Rx trip due to reactivity insertion |
| 3. LOPCS | Rx trip due to loss of PCS flow |
| 4. LOSCS | Rx trip due to loss of SCS flow |
| 5. LOCA-I | Rx trip due to loss of coolant (PCS pipe break- out pool) |
| 6. LOCA-II | Rx trip due to loss of coolant (PCS pipe break- in pool) |
| 7. LOCA-III | Rx trip due to loss of coolant through beam tube |
| 8. GTRN | Rx trip due to all other transients |

## 4. Event Tree Analysis

Event trees are developed for the eight selected initiating events.

### 4.1. Loss of Electric Power (LOEP)

An event tree for LOEP represents the possible responses of the reactor to a loss of normal electric power as shown in Fig. 3. A loss of normal electric power is initiated by a loss of off-site grid power. If a LOEP occurs, primary cooling pumps, secondary cooling pumps and cooling tower blowers come to stop. As soon as the electrical power supply to the reactor shutdown system is cut off, the reactor power decreases rapidly by the immediate insertions of Control Absorber Rods (CARs) and Second Shutdown Rods (SSRs). At the loss of normal electric power, a diesel generator starts to supply power to one PCS pump in train B. Following a reactor trip, PCS pump in train B starts automatically to perform the pool cooling. If it failed, the flap valve or siphon break valve should open to establish natural circulation through the pool water, which serves as a huge heat sink.

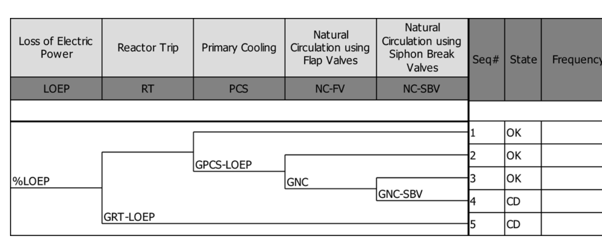


Fig. 3. Event tree for LOEP.

### 4.2. Reactivity Insertion Accident (RIA)

An event tree for RIA models the possible responses of the reactor to reactivity insertion events as shown in Fig.

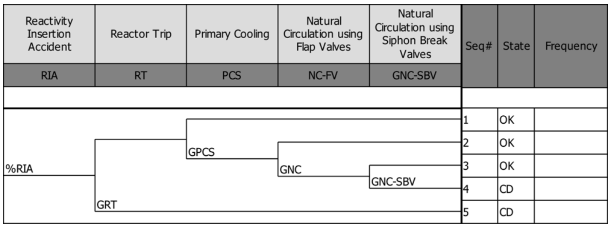


Fig. 4. Event tree for RIA.

4. If an excess reactivity is inserted, the reactor core power abnormally increases and may result in a damage of the fuel plates. An RIA can be initiated by an event such as an inadvertent ejection of control rod due to operator errors or a failure of Control Rod Drive Mechanism (CRDM) or the reactor regulating system during normal operation. If the reactor trip by the RPS or APS succeeds, the reactor core will be cooled by the PCS pumps or the pool water natural circulation through flap valves or siphon break valves.

## 4.3. Loss of PCS Flow (LOPCS)

An event tree for a loss of PCS flow models the possible responses of the reactor to a loss of PCS flow as shown in Fig. 5. A loss of PCS flow can occur when the PCS pumps are malfunctioning owing to the power supply or mechanical problem. The PCS flow can also be reduced when the paths are partially blocked owing to a valve closure or blockage of foreign objects. This initiating event is mainly caused by the instantaneous failure of two PCS pumps, the failure of one PCS pump or coolant reduction. The reactor will be tripped by either a low PCS flow signal or low core Differential Pressure (DP) signal. If the reactor trip by the RPS or APS succeeds, the reactor core will be cooled by the natural circulation through flap valves or siphon break valves.

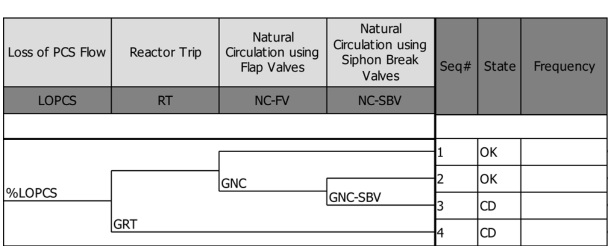


Fig. 5. Event tree for LOPCS.

## 4.4. Loss of SCS Flow (LOSCS)

An event tree for a loss of SCS flow models the possible responses of the reactor to a loss of SCS flow as shown

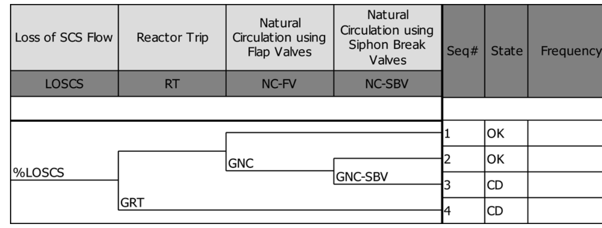


Fig. 6. Event tree for LOSCS.

in Fig. 6. A loss of SCS flow event considers a loss of cooling by the secondary cooling system during normal power operation. A loss of secondary cooling flow can be caused by failures of the secondary cooling pumps or valves, a rupture of the piping, a blockage of the flow path, and a failure of the cooling towers. After the reactor is shutdown, the decay heat is removed by the natural circulation of the pool water through the flap valves or siphon break valves.

## 4.5. Loss of Coolant outside the Pool (LOCA-I)

An event tree for LOCA-I models the possible responses of the reactor to a loss of coolant event outside the reactor pool, as shown in Fig. 7. For ruptures of the coolant pipe outside the reactor pool, the reactor coolant spills out to the reactor hall through the break location, and subsequently the pool water level and the PCS flow decrease, but the core differential pressure increases owing to the core flow increase. When the pool water level reaches a specified level, the siphon break valves begin to open. Then air is sucked into the reactor outlet PCS pipe, and the discharge flow then comes to decrease. Finally, the discharge flow stops completely before the pool water level reaches a prescribed water level. After the siphon flow is blocked, a natural circulation flow through the flap valves, the reactor outlet PCS pipe, the core outlet plenum, the reactor core, the upper guide structure, and the reactor pool is established.

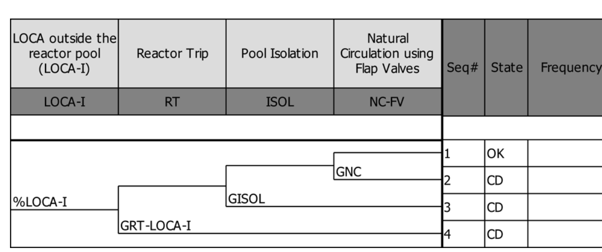


Fig. 7. Event tree for LOCA-I.

## 4.6. Loss of Coolant inside the Pool (LOCA-II)

An event tree for LOCA-II models the possible response

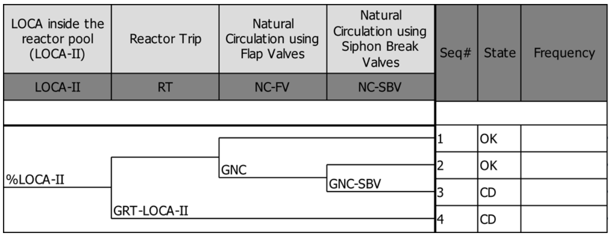

Fig. 8. Event tree for LOCA-II.

of the reactor to a loss of coolant inside the reactor pool, as shown in Fig. 8. For ruptures of the reactor outlet PCS pipe inside the reactor pool, the PCS pumps draw the pool water through the break position and the reactor core so that the core flow is reduced to a certain level. The core flow reduction depends on the break size. At an early stage of the event, the core decay heat will be removed by the reduced core flow. After PCS pumps are turned off, the core flow direction is switched from downward to upward by the buoyancy force and the core decay heat is then removed by a natural convection flow through the flap valves or siphon break valves.

### 4.7. Beam Tube LOCA (LOCA-III)

An event tree for a loss of coolant due to beam tube rupture models the possible response of the reactor to a loss of coolant owing to beam tube rupture as shown in Fig. 9. This accident is caused by a simultaneous break of both a beam tube in the reactor pool and a beam port flange at the end of the beam port during normal operation. If a beam tube and a beam port flange break down at the same time, the reactor pool water is discharged to the reactor building through the broken beam port assembly. At this stage, the reactor core is covered with emergency water fed into the core outlet pipe.

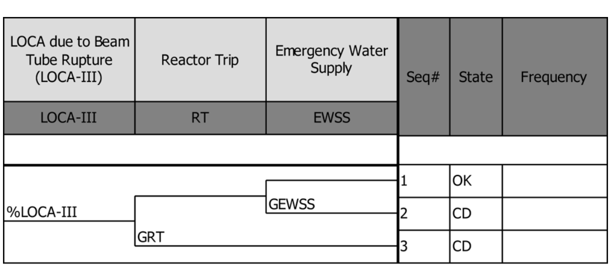

Fig. 9. Event tree for LOCA-III.

### 4.8. General Transients (GTRN)

An event tree for GTRN models the possible response of the reactor to general transients as shown in Fig. 10.

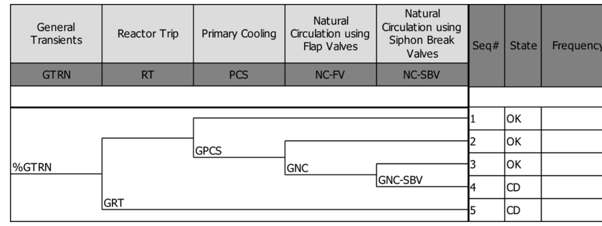

Fig. 10. Event tree for GTRN.

GTRN involves a diverse group of non-LOCA and non-accident initiating events in which a process parameter perturbation leads to a reactor trip. If the reactor trip by the RPS or APS succeeds, the reactor core should be cooled by the PCS pumps or the natural circulation using flap valves or siphon break valves.

## 5. Fault Tree Analysis

Simple fault tree models for a PSA are developed instead of a detailed model at this conceptual design stage. The failures of the major components and dependencies between systems have been considered for a fault tree analysis. Normal operating trains were assumed to have a pump, a check valve and a manual valve. The failures of the pumps and support systems such as the electrical power are modeled, and the failure of the check valve or manual valve is also modeled for the train. Of course, the common cause failure (CCF) and operator error are modeled. Table 3 shows the system modeled in the fault tree analysis and the modeling descriptions by systems.

Table 3. System modeling

| System | Modeling |
|---|---|
| Primary cooling system | 2 top logics are developed for 1) a failure of decay heat removal mode and 2) a failure of standby PCS train. CCFs, operator error and electric power are modeled. |
| Isolation of broken PCS pipe | The mechanical and signal failures of siphon break valves are modeled. |
| Natural circulation in reactor Pool | The failures of flap valves and siphon break valves are considered.<br>Valves and CCF are modeled. |
| RPS & APS | Three major functions are modeled for the RPS (Reactor Protection System), which are the mechanical failure of the control rods, the failure of the trip relays and the failure of trip signal.<br>The APS (Alternate Protection System) is also modeled like the RPS. |
| Emergency water supply system | The failures of emergency water injection valves, their signal and electric power are considered. Also, the failure of level transmitters and their CCF are modeled. |
| Electrical power system | The main power supply from the offsite grid and emergency diesel generators are considered. The failure of 460V AC or 125V DC are modeled in detail, which are modeled as a part of 4.16 KV bus. Also, the failure of transformer or circuit breaker between buses is considered. |

## 6. Data Analysis

A research reactor is in the design stage and has no operating experience. Therefore, no plant specific failure data are available. Thus the data used in the PSA are based entirely on generic failure rates. The major generic sources are IAEA-TECDOC-930 Generic Component Reliability Data for Research Reactor PSA[4] and ALWR PRA Key Assumptions and Groundrules (KAG) of revision 7[8]. However, the generic database was compared with several sources during its development. The following sections describe the details of the data analysis.

### 6.1. Initiating Event Frequencies

This section discusses initiating event frequencies and provides the details for obtaining each initiating event frequencies and its data source. The initiating events used in the PSA were identified and listed in section 3. The quantification of the initiating event frequency model was based generally on generic initiator frequency data. In the PSA, the initiating event frequencies used are based on those used in the HANARO research reactor PSA[9]. The results of the initiating event frequencies are summarized in Table 4.

Table 4. Initiating event frequencies

| IE | Description | Frequencies (/yr) |
|---|---|---|
| LOEP | Loss of normal electric power | 3.13E+00 |
| RIA | Reactivity insertion accident | 1.70E-02 |
| LOPCS | Loss of PCS flow | 4.03E-01 |
| LOSCS | Loss of SCS flow | 9.46E-02 |
| LOCA-I | LOCA outside the reactor pool | 6.60E-03 |
| LOCA-II | LOCA inside the reactor pool | 1.52E-03 |
| LOCA-III | LOCA due to beam tube rupture | 1.02E-07 |
| GTRN | General transients | 3.00E+00 |

### 6.2. Component Reliability Data

The reliability data used in the preliminary analysis are presented in Table 5. Component hardware failures imply failures of the components to function as required owing to internal defects. The common cause failure probabilities are considered. Because this research reactor at the design stage has no plant specific failure experience data, failure probabilities presented herein were assessed based on generic data sources[4,8,9].

Table 5. Component reliability data

| Description | Data used | Ref. |
|---|---|---|
| Bus failure | 2.30E-06/h | [1] |
| Trip relay | 1.00E-04 | [2] |
| Diesel generator fails to run | 7.02E-04/h | [1] |
| Diesel generator fails to start | 3.00E-03 | [1] |
| Flap valve fails to open | 5.00E-04 | [1] |
| Circuit breaker fails to close on demand | 1.00E-03 | [1] |
| Motor driven pump fails to run | 2.19E-06/h | [1] |
| Motor driven pump fails to start | 1.00E-03 | [1] |
| Motor operated valve fails to open | 1.00E-04 | [1] |
| Siphon break valve fails to open | 4.50E-06/h | [1] |
| Heat exchanger fails while operating | 1.07E-05/h | [1] |

[1] IAEA, Generic Component Reliability Data for Research Reactor PSA, IAEA-TECDOC-930, 1997.
[2] ALWR PRA Key Assumptions and Ground Rules, Rev.7, EPRI, 1997.

### 6.3. Common Cause Failure Data

The Alpha-factor method is used to model the Common Cause Failure (CCF) events. For components for which no experience data are available, alpha-factors are assumed based on the generic values presented in KAERI/TR-2916/2005[10].

### 6.4. Human Error Probability Data

Post-initiator events are human errors made in response to the mitigation of an initiating event. This type of errors occur during a situation assessment or task execution after an accident, and are related with operator actions performed in response to an Emergency Operating Procedure (EOP) or recovery actions to resolve a failed safety function. The probability of post-initiator events was concerned with the available time to diagnose the event, the location where the task is performed, the familiarity level to the situation, the type of cognitive processing associated with the task, and the stress level. Three post-initiator events were identified in this HRA. The quantification of post-initiator events was conducted basically based on the ASEP HRA procedure[11]. A conservative screening value of the Human Error Probability (HEP) was used for post-initiator events in the preliminary PSA.

## 7. Accident Sequence Quantification

Because the CDF by each initiating event is quantified by the sum of all core damage accident sequences contained in each event, this section provides only a summary of their results and a description of the findings.

A total of 32 core damage accident sequences for an internal event analysis were identified and quantified using the AIMS-PSA5). Of them, only 16 sequences were included in the CDF model for internal events. The criterion for inclusion was all sequences with a point estimate frequency greater than a truncation value of 1.0E-13/yr.

The summary of results are shown in Table 6, including the results of the contributions to the total CDF by initiating events. In addition, Fig. 11 represents the contribution to the total CDF of each initiating event by a phi chart. LOCA-I has a dominant contribution to the total CDF by a single initiating event. The preliminary quantification results indicate a point estimate of 7.38E-06/yr for the overall CDF attributable to internal initiating events for a research reactor.

The contributions of initiating events and the findings can be characterized as follows:

(1) LOCA-I has a dominant contribution to the total CDF by a single initiating event.
(2) LOCA-I provides 73.8% of the total CDF. The most dominant contributor is a Common Cause Failure (CCF) of siphon break valves. This result is due to the credit of the valve recovery action performed, particularly in the sequence of LOCA-I.
(3) GTRN has the second largest contributor at 16.8% due to a failure of reactor trip using RPS or APS.
(4) The third important initiating event is LOCA-II, providing 5.8% of the total CDF. This is due to a failure of reactor trip using RPS or APS.
(5) The contributions of LOPCS, LOCA-III, LOSCS, RIA and LOEP are relatively small, although they are the next significant contributors. These initiators contribute less than 2.5% of the total CDF.

The dominant MCSs are described below.

• LOCA-I * PCLVW-SV0102

Following a LOCA-I, this scenario happens when two siphon break valves do not work simultaneously owing to the same mechanical cause. The dominant contributor to the CDF of this sequence is the CCF of the siphon break valves. The estimated CDF of this sequence is about 3.71E-07/yr as a point estimate value, and this MCS provides 50.3% of the CDF.

• LOCA-I * PCCVW-FLAPV

Following a LOCA-I, this scenario happens when two flap valves do not work simultaneously owing to the same mechanical cause. The dominant contributor to the CDF of this sequence is the CCF of the flap valves. The estimated CDF of this sequence is about 8.51E-08/yr as a point estimate value, and this MCS provides 11.5% of the CDF.

Table 6. Core Damage Frequencies

| Initiating Event | IE Frequency (/yr) | CDF | % |
|---|---|---|---|
| LOCA-I | 6.60E-03 | 5.45E-07 | 73.8 |
| GTRN | 3.00E+00 | 1.24E-07 | 16.8 |
| LOCA-II | 1.52E-03 | 4.29E-08 | 5.8 |
| LOPCS | 4.03E-01 | 1.70E-08 | 2.3 |
| LOCA-III | 1.02E-07 | 4.77E-09 | 0.6 |
| LOSCS | 9.46E-02 | 3.98E-09 | 0.5 |
| RIA | 1.70E-02 | 7.01E-10 | 0.1 |
| LOEP | 3.00E+00 | 7.35E-11 | 0.0 |
| Total | | 7.38E-07 | 100.0 |

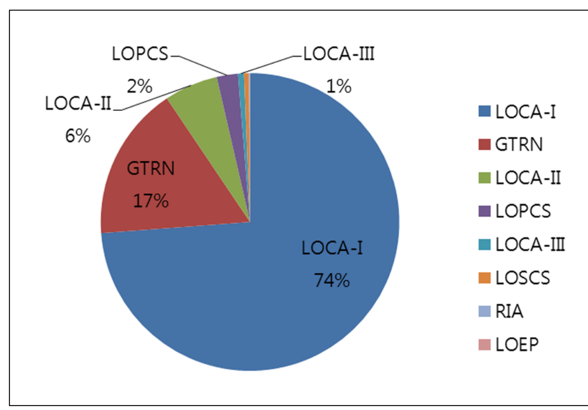


Fig. 11. Initiating Event Contribution to CDF.

## 8. Conclusions

This preliminary PSA was undertaken to assess the level of safety for the design of a research reactor and to evaluate whether it is probabilistically safe to operate and reliable to use. The current design was improved more safely through the PSA sensitivity study. The title of sensitivity study paper is 'Design Improvement to a Research Reactor for Safety Enhancement Using PSA', which is scheduled to be published in the journal of the Korean Society of Safety

The principal conclusions from this study are as follows:

• The CDF for the representative initiating events is less than 1.0E-6/year even though conservative assumptions are used in the reliability data. The conceptual design of this research reactor is designed to be sufficiently safe from the view point of safety.

- In the preliminary PSA, the initiating event frequencies used are based on those used in the HANARO research reactor PSA.
  - LOCA frequencies are much higher than that of other research reactors, and therefore LOCA-I has a dominant contribution to the total CDF by a single initiating event.
  - A more realistic LOCA frequency will be used in the final PSA and the CDF of the final PSA will be expected to be less than that of the preliminary PSA.
- In the final stage of the PSA, a detailed analysis will be conducted including the external events, to estimate the entire risk of this research reactor, and to confirm the assumptions made. If any deficiency in the safety is found during the final PSA, feedback on design will be made to meet the employer's safety goals and requirements.
- The PSA methodology is very effective to improve reactor safety in a conceptual design phase and especially, RID is very nice way to find the deficiencies of a reactor under design and to improve the reactor safety by solving them

**Acknowledgments:** This work was supported by Nuclear Research & Development Program of the National Research Foundation of Korea (NRF) grant, funded by the Korean government, Ministry of Science, ICT. (No. 2017M2A8A 4015287).

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