

4<sup>th</sup> Virtual Annual Meeting on ASEAN Network on Nuclear Power Safety Research

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ASEAN NPSR

Discussion on Nuclear Safety

Research related to Research Reactor

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# Interest in Nuclear Safety Research related to Research Reactor (RR)

## Probabilistic Risk Assessment (PRA)

- **Collection of data**
- **Hazard analysis**
- **Cause and consequence analysis**
- **Initiating Events (IEs) Identification**
- **Risk quantification of IEs**
- **Discussion on the overall risk of RRs**



# Collaborative Networks

## ASEAN NPSR

ASEAN Network on Nuclear Power Safety Research



**ASEAN**  
Large Nuclear  
and Synchrotron  
Facility Network



## ASEAN NPSR

# Networks' Frameworks

## ASEAN LNSN framework

- **Research Reactor** is recognized only one nuclear facility type in ASEAN region for research sharing with each other.

## ASEAN NPSR framework

- **Risk of nuclear consequences from Research Reactor in ASEAN region is concerned** about the nuclear accident and consequences as in NPP accidents in the past.



# Overall PSA of RRs

- **Research Reactor is recognized only one nuclear facility type in ASEAN region using fission reaction to generate power.**

ASEAN MSs	Reactor Names	Reactor Types	Power	First operation
Thailand	Thai Research Reactor 1/Modification 1 (TRR-1/M1)	TRIGA-Mark III	1.2 MW	1962 (59 yr.)
Vietnam	Dalat nuclear research reactor	TRIGA-Mark II	250 kW → 500 kW	1962 (59 yr.)
Malaysia	PUSPATI TRIGA Reactor (RTP)	TRIGA-Mark II	1 MW	1982 (39 yr.)
Philippine	Philippine Research Reactor-1 (PRR-1)	TRIGA-converted reactor	1MW → 3 MW	1963 (51 yr.)
Indonesia	TRIGA MARK II Research Reactor in Bandung	TRIGA-Mark II	250 kW → 2 MW	1965 (49 yr.)
	Kartini nuclear research reactor	TRIGA-Mark II	100 kW	1979 (42 yr.)
	MPR RSG-GA Siwabessy nuclear research reactor	Multi-Purpose Reactor	30MW	1987 (34 yr.)



Level 1 PRA of Research Reactors in ASEAN

Internal event

External event

Human failure

Hazard analysis

Cause and consequence analysis

Initiating event identification

Risk Quantification

Discussion on the overall risk of RRs in ASEAN

- Plant survey
- Expert judgment

- Plant-specific data
- Generic data
- Statistic analysis

# Starting points of PSA of RRs in ASEAN

## Level 1 PRA of Research Reactors (RRs) in ASEAN

- Three Levels of PSA is considered based on general PSA levels of the NPPs based on Light Water Reactors (LWRs).
- **Graded approach** of IAEA was applied to simplify the safety requirements consistently the existing circumstances, regulatory and management systems.
- **Level 1 PSA** could be the first possible main focus for RRs based on the graded approach because Structures, Systems, and Components (SSCs) of reactor are significantly less complicated than that of the normal LWRs.



# Starting points of PSA of RRs in ASEAN

## Level 1 PRA of RRs: Internal Events

- Level 1 PRA was planned to quantify Core Damage Frequency (CDF) from the three event-types consisting of **internal events, human errors, and external events**.
- Due to enough publicly available data or existing **generic data** (International Atomic Energy Agency, 1997), **internal events** may be a potential assessment to conduct PSA for beginners in this field.



# Starting points of PSA of RRs in ASEAN

## Level 1 PRA of RRs: Plant specific data

- However, in the PSA, **plant specific data** is still the important part that can represent the risk of RRs consistently existing RRs more.
- Plant specific data can be collected and surveyed based on **the record data and experiences of operators**.
- Many old RRs, that has never been conducted PSA before, may face the lack of information and a limited number of maintenance data for each equipment to provide the **plant specific data for PSA**.



# Starting points of PSA of RRs in ASEAN

## Level 1 PRA of RRs: Plant specific data

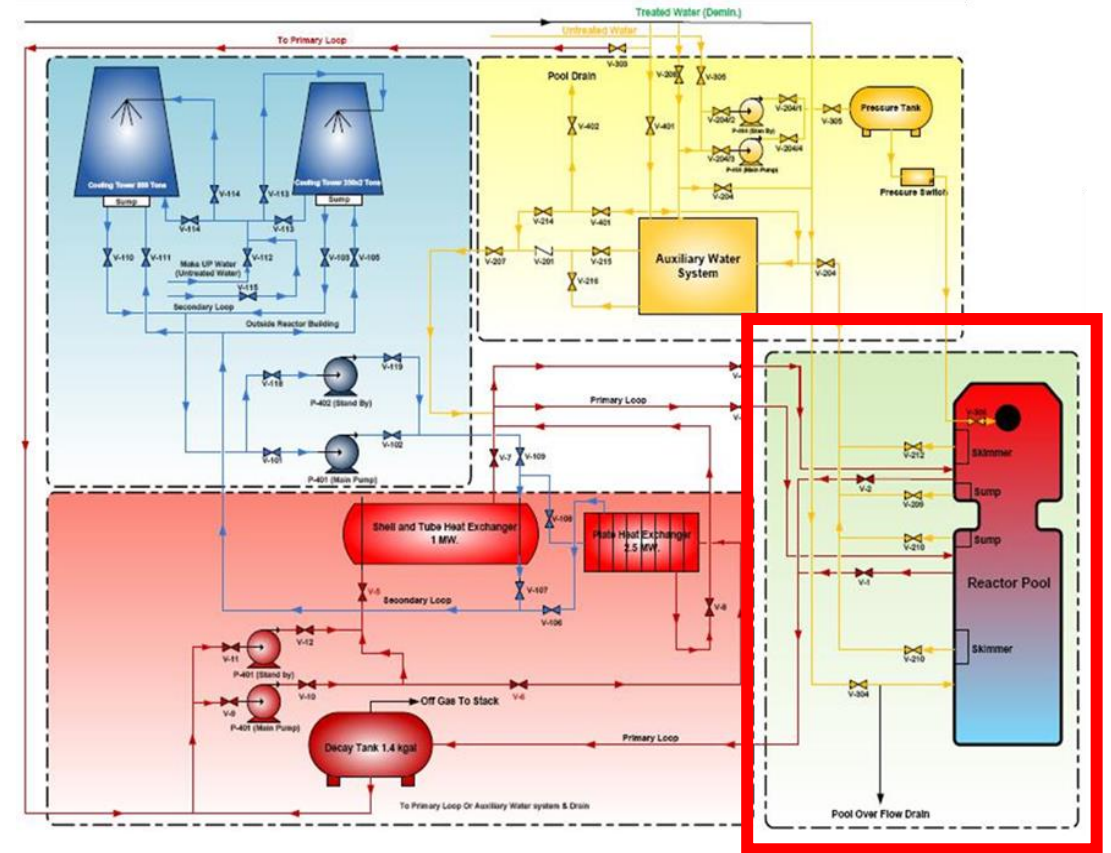
- To contribute to risk assessment of RRs in ASEAN region, the collection of **plant specific data for existing RRs** in ASEAN may be **a first potential activities** that we can work to have the proper data base of RRs in ASEAN together consistently the Safety Standard of RRs.
  - **Data records** → extract from logbook/ interview senior operators
  - **Plant survey** → confirm the results of failure data or hazard analysis



# Example: Plant survey of TRR-1/M1

IE Group No.	HAZOP Analysis
1	Loss of coolant accident (LOCA)
2	Insertion of excess reactivity
3	Erroneous failure of equipment or components
4	Loss of flow accident (LOFA)
5	Loss of electrical power supplies

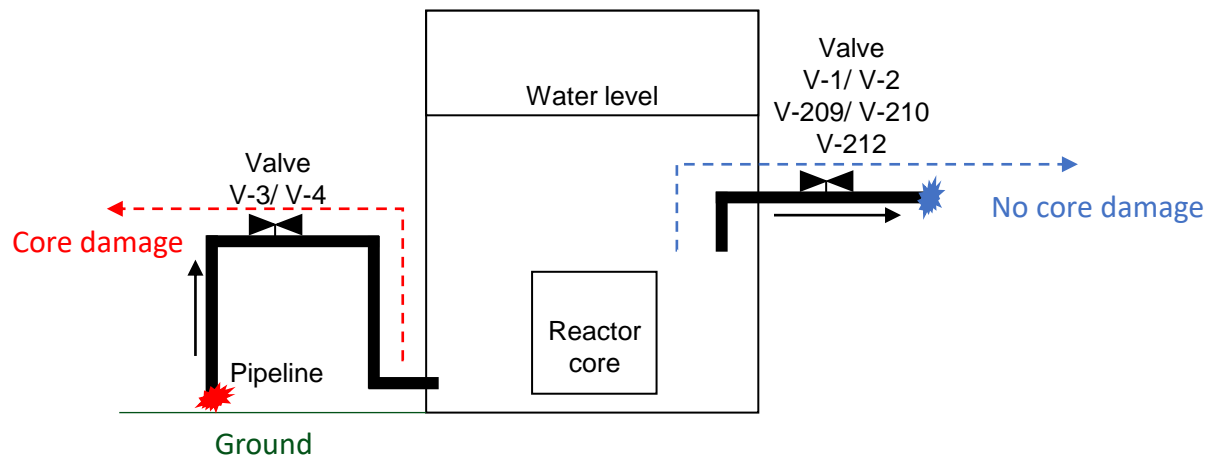
- During hazards analysis of TRR-1/M1 based on cooling system scheme with no consideration of height, all of **outlet flow valves** are considered in case of outlet flow valves failed open.
- After surveying of TRR-1/M1 plant, it was found that all of outlet flow valves are above water level in the reactor core.
- LOCA of all of outlet flow valves does not trigger core damage.



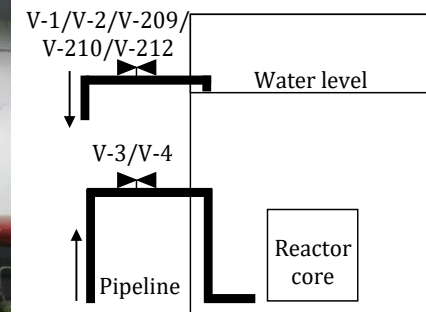
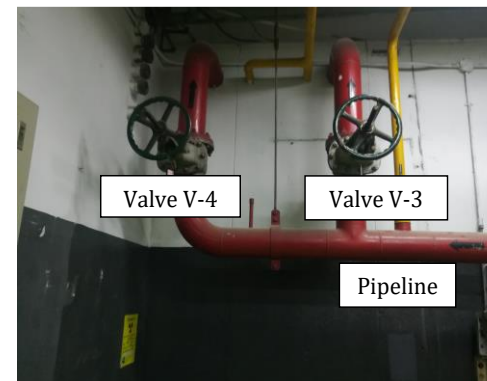
Reactor Cooling System (RCS)

# Example: Plant survey of TRR-1/M1

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- However, we found that inlet flow valves of Valves V-3 and V-4, that we has not considered before, are below the water level.
- Valves V-3 or V-4 failed open together with pipe break are considered to trigger LOCA.
- Not only the consideration of the design and schemes, but also thorough plant survey is necessary when conducting the hazard analysis of PRA.



# Activities

## Level 1 PRA of RRs: Plant specific data

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# ASEAN NPSR

Let's Discuss

