ASPECTS OF PROBABILISTIC SAFETY ASSESSMENT FOR TRIGA RESEARCH REACTOR

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The main aim in the development of probabilistic safety assessment (PSA) is to assure the safety and reliability of the installations in the design, construction or operation. In regard to PSA, research reactors have several peculiarities; they are more simple installations than the nuclear power plants, power generated is less than in the nuclear power plants (NPPs) and require more flexibility in operation due to experimental devices, research and irradiation facilities. Consequently, human actions in the operation and maintenance of the RRs are also specific, due to multiple arrangements of the core and work for experiments.

The paper focuses on the PSA model for Romania TRIGA Steady State Reactor 14 MW with all the aspects pertaining to PSA: initiating events, event trees, fault trees, accident sequences, reliability data, etc.

Keywords: PSA, TRIGA research reactor

1. Introduction

A huge volume of information and experience has been accumulated in the last decades through application of PSA techniques to nuclear power reactors. The general information and PSA evaluation methodology in particular can be very useful for analysts evaluating PSA for research reactors.

Many of the research reactors have long operational lives and a diversity of associated experimental facilities. These undergo an ageing process and obsolescence, requiring consideration of refurbishment at a reasonable cost. RRs are generally, more simple, with fewer systems and, accordingly, more easy to analyze than NPPs. Actually, there are more than 300 research reactors around the world, with a diversity of constructive types and thermal power ranges, from the few watts up to hundred megawatts.

International Atomic Energy Agency (IAEA) has promoted and sustained the developmant of PSA for research reactors in the Member States. Some of these reactors are assisted by IAEA in the framework of technical projects, IAEA being directly involved in their safety evaluation. Since the late eighties, the International Atomic Energy Agency (IAEA) undertook a Co-ordinated Research Programme (CRP) on PSA for Research Reactors [1] which helped promote use

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of PSA for this kind of nuclear facilities. Modern projects [2] for research reactors include PSA evaluations for the postulated accident sequences while the requirements of national regulatory boards concerning the design and operation of research reactors have been continuously extended and refined as a result of international practices and recommendations.

Throughout early 2005, an adequate Information System structure as well as software based query tools were developed by JRC in order to enable "qualified users" to detect in an easily understandable way the main differences and similarities of safety approaches in different RR facilities. The Information System offers the possibility to exchange experience in the area among interested parties and could thus represent a starting point for future harmonization of RR safety principles. This Information System, called DARES (DAtabase for REsearch Reactor Safety), is installed on the JRC's ODIN website http://odin.jrc.nl. In total, 30 RR facilities from Europe, Argentina, Australia, Canada, South Africa and the USA sent so far information to DARES.

In Romania, PSA is applied for Cernavodă NPP. The first PSA project for Cernavodă NPP was started in 1987 by the PSA group at the Institute for Nuclear Research (INR). For TRIGA steady-state reactor operated by INR, the PSA project began after 2002. The paper presents a series of aspects and results of the TRIGA PSA project.

2. Probabilistic Safety Analysis for TRIGA research reactor

In what follows a short description of TRIGA reactor is offered together with the scope and specific elements of the PSA analysis. These elements include the initiating events (IEs), the description of the final core damage states and radioactive release categories, event trees (ETs), fault trees (FTs) and reliability data used for FTs.

2.1 Short description of TRIGA reactor

Romania TRIGA reactor was commissioned in 1980 (first criticality was reached on November 17th 1979) the producer being the General Atomic Company from US. The dual-core concept involves the operation of TRIGA high-flux, steady-state research and materials testing reactor at one end of a large pool, and the independent operation of an annular-core pulsing reactor (TRIGA-ACPR) at the other end of the pool.

The steady-state reactor (SSR) was intended for long-term testing of power reactor fuel components (pellets, pins, subassemblies and fuel assemblies), isotopes production for medical use and neutron experimental physics techniques.

The annular core pulsing reactor (ACPR) is used for transient testing of power reactor fuel specimens. Both reactors are supplied with beam tubes. Both reactors may be operated separately or at the same time. The two reactors are independent of each other with two exceptions: they share a common reactor pool and same cooling and water purification systems [3].

SSR is a forced convection reactor cooled via a primary circuit with 4 pumps and 3 heat exchangers. The ACPR is natural convection reactor cooled by the pool water.

2.2 Scope of PSA analysis for TRIGA reactor

The purposes of the PSA analysis for TRIGA SSR are:

- Treatment of internal and external IEs;
- Evaluation and calculation of sequences that are leading to fuel failure and fission product occurrence.

The above mentioned topics pertain to Level 1 and 2 PSA.

We start the analysis from the following premises:

- Only reactor was considered as possible source of radioactive releases. Although, the fuel failure in the irradiation devices, designed for this event may represent a radiological risk for the operating personal if particular safety barriers are inefficient, this failure is not considered as a final state in the event trees. In our analysis the events due to experimental devices for TRIGA reactor were not credited.
- Operation at maximum power (14 MW) is considered as being the bounding case in risk assessment, and PSA evaluations has been made for this situation.
- Reactor fuel was considered damaged when the fuel temperature limit is exceeded, according to the TRIGA Final Safety Report [3] limits.
- Quantification analysis was performed using specific reliability data of TRIGA SSR reactor, fruit of collection and processing of raw data for obtaining reliability data, but also generic data taken from IAEA available sources for research reactors [4].
- For quantification of human errors we considered calculated values using OAT (Operator Action Tree) [5] or THERP [6].
- The analysis of source term, inventory and transport of fission products in the primary circuit and in the reactor hall, as support of Level 2 PSA is not treated in this paper.

2.3 Initiating Events for TRIGA SSR 14 MW reactor

This chapter presents the possible initiating events (IEs) for TRIGA SSR 14 MW reactor based on Safety Analysis Report, deterministic analysis and

initiating event list considered by IAEA for research reactors. Initiating event frequency both with method used for calculation is included in the Table 1.
Generally, there are three main methods to evaluate the frequencies of Initiating Events:

- Method 1. Combining the reactor operation experience with generic data of the frequencies initiating events. This method is used frequently due to absence of specific data.
- Method 2. Use the data referring to the initiating events frequencies based on expert judgments.
- Method 3. The evaluation of initiating events through fault tree method, generally used in case of loss of support systems.

A series of IEs were excluded from further evaluation due to "cut- off" criterion based on the small values of calculated frequency (in case of LOCA2 I/E). Others were not taking into account further due to dedicated deterministic analysis (CHDL I/E) or Safety Report of TRIGA reactor (FCB I/E). In case of earthquake, we perform a qualitative analysis.

Further PSA analysis takes into account LOFSP I/E, LOFA I/E, LOCA1 I/E.

Table 1

Postulated initiating events, method of calculation, frequency for TRIGA SSR 14 MW reactor

Initiating Event	**Method of calculation**	**Frequency (occ./year)**
Loss of power supply – LOFPS I/E	Fault Tree Analysis	7.88E-03
Criticality during handling (fuel insertion error) – CHDL I/E	Human Error Analysis (TESEO method) + Operating experience	2.1E-02
Loss of flow (failure of Primary Pumps Lines) – LOFA I/E	Fault Tree Analysis	1.74E-01
Fuel Channel Blockage – FCB I/E	TESEO + Maintenance Requirements	7.5E-03
Loss of coolant accident (Primary Pipe Rupture) - LOCA1 I/E	Formula for Steel Pipes Rupture (Thomas)	1.E-02
Loss of coolant accident through transfer gate failure followed by beam tube rupture - LOCA2 I/E	Fault Tree Analysis	**1.67E-09**
Earthquake	Safety Analysis of the Romanian TRIGA facility designer	1.E-04

2.4 Description of final core damage states for TRIGA reactor

According to [8], based on thermohydraulic analysis, only three final core damage states D1, D2, D3 were considered. Table 2 includes the percent of damaged core, mean frequencies and statistical confidence intervals limits (5%, 95%).

It can be seen in Table 2 that not all the identified sequences are leading to the same core damage degree of TRIGA reactor. The highest contribution (about 100 %) is due to D1 state, failure of 725 fuel elements in water.

Table 2

Final core damage states for TRIGA reactor

Final core damage state	**Percent of damaged core**	**Mean frequency / year**	**Confidence interval 5% / year**	**Confidence interval 95% / year**
D1	100%, in water	7.22E-06	6.68E-06	1.01E-05
D2	80%, in air	3.98E-15	2.73E-16	2.31E-14
D3	100%, in air	2.09E-15	1.49E-16	1.15E-14

The main contribution leading to D1 state is due to combination of initiating event, LOFA, common cause failure of control rod mechanisms or common cause failure of control rods.

2.5 Radioactive release categories for TRIGA reactor

Radioactive release categories (see Table 3) depend on release quantity in the reactor hall, on hall isolation and state of radioactive products disposal system (emergency ventilation system).

Three reactor hall states are considered (see Table 4), depending on assurance of reactor hall isolation and availability on emergency ventilation and its associated air filters. These states combined with the three core damages states for TRIGA reactor produce nine categories of radioactive releases.

Table 3

Radioactive release categories

Release category	**Core damage state**	**Containment state (reactor hall)**
R1	D1	C1
R2	D1	C2
R3	D1	C3
R5	D2	C1
R6	D2	C2
R7	D2	C3
R8	D3	C1
R9	D3	C2
R10	D3	C3

Table 4

Containment states (reactor hall)

State index	Reactor hall isolation available	Emergency ventilation available
C1	Yes	Yes
C2	Yes	No
C3	No	-

The radioactive releases states and its associated corresponding frequencies and confidence interval limits are given in the Table 5. One can note that the radioactive release state, R1, representing 99.5%, is dominant, followed by R2 state with a 0.0047% contribution.

Contribution to R1 state is given by initiating event LOFA and combination of common cause failures of control rod and control rod mechanisms.

Table 5

Frequencies and confidence interval limits for radioactive release states

Radioactive release category	Mean frequency / year	Confidence interval 5% / year	Confidence interval 95% / year
R1	7.22E-06	6.69E-06	1.09E-05
R2	4.39E-08	3.89E-08	7.71E-08
R3	2.54E-10	3.60E-11	7.20E-10
R5	3.98E-15	2.55E-16	2.28E-14
R6	1.81E-17	1.35E-18	1.15E-16
R7	1.69E-19	5.13E-21	1.25E-18
R8	2.09E-15	1.52E-16	9.87E-15
R9	9.45E-18	7.28E-19	6.80E-17
R10	8.90E-20	2.13E-21	5.54E-19

2.6 TRIGA reactor event trees

Some general assumptions were considered in the evaluation of event trees. These are referring to:

- Unavailability of emergency cooling does not lead to fuel damage. Actually, as shown the commissioning tests in 1979, main pumps inertia and natural convection loop established after reverse flow are able to remove residual heat from a 14 MW reactor initial power without the emergency pump. Temperature evolution in the hottest pins did not significantly rose during the experiment with emergency pump stopped from the beginning of the test [7]. More than that, from the safety analysis point of view, emergency loop is evaluated and is able to remove heat for a "coast-down" time more than 2 seconds in a scenario in which scram initiation appears in 0.2 seconds after scram signal produced after a flow decrease [3].

- The *natural convection* is done automatically through unheated water zones, with no action from any reactor active components (as an example natural convection valves that exist at other research reactors).
- The *secondary cooling system* was not considered in the accident sequences. This is because the transient duration which is leading to fuel damage is small, the global transfer from primary to secondary being unimportant. Also, the residual heat after successful operation of reactor automatic or manual shutdown may be absorbed by the big volume of pool water and primary circuit.

Below is presented the event tree corresponding to LOFA initiating event, its contribution being significant to core damage and fission product occurrence.

Based on deterministic analysis [8], loss of flow initiating event consist in failure of 2 of 2 main circulating pumps. If primary cooling is lost (main pumps are unavailable) reactor has the opportunity to cool the reactor via emergency circuit for fuel heat removal and heat produced after reactor scram.

Deterministic analyses are necessary to obtain the fuel temperature evolution when scram initiation does not follow the flow decrease that is automatic shutdown system fails. Following the course of the events it appears the necessity to shutdown the reactor by reactor operator.

Fig. 1 presents the ET for the loss of forced circulation IE. The ET considers fission product release states only in the sequences when the reactor automatic shutdown system fails on flow channels, difference inlet outlet pool water temperature channels, fuel temperature channels, but also, unsuccess of manual scram. If it takes into account the reactor design and regulatory requirements which stipulate the mandatory practice of emergency cooling for residual heat removal, we should consider that the failure of emergency cooling immediately after the successful operation of automatic shutdown (after 14 MW functioning) on one or another scram channels mentioned above, will lead to fuel damage states. Actually, as demonstrated by commissioning test mentioned above**,** failure of emergency cooling does not lead to fuel damage.

Following the event tree, in case of unsuccessful operation of automatic shutdown, the reactor shutdown using operator action to act scram actuator is tested.

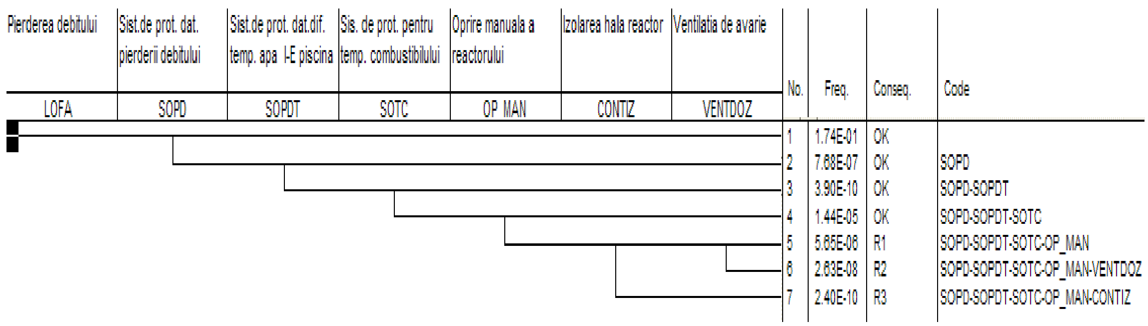

Fig. 1. Event tree/ accident sequences for LOFA initiating event

2.7 Evaluation of headings/ systems in the event trees (ET)

The headings corresponding to event trees for TRIGA model that are arising from system failures, human errors, etc, were evaluated using fault tree method. Table 6 presents the corresponding headings for LOFA, ET and unavailability values and its associated confidence interval limits.

Table 6

Unavailabilities and associated confidence interval limits for LOFA event tree headings

Heading index	Heading description	Unavailability	5% limit	95% limit
SOPD	Reactor shutdown system due to loss of flow fails to function	1.19E-04	1.16E-04	1.25E-04
SOPDT	Reactor shutdown system on difference inlet outlet pool water temperature fails to function	1.19E-04	1.16E-04	1.27E-04
SOTC	Reactor shutdown system on fuel temperature fails to function	1.16E-04	1.14E-04	1.20E-04
OP_MAN	Reactor manual shutdown fails to function	1.02E-02	1.02E-02	1.02E-02
CONTIZ	Reactor hall isolation fails to function	4.25E-05	5.92E-06	1.30E-04
VENTDOZ	Emergency ventilation system fails to function	4.65E-03	4.63E-03	4.69E-03

2.8 Reliability data used in the fault trees

In performing level 1 and 2 PSA (event trees, fault trees, accident sequences) we used in the beginning PSAMAN code ([9], [10]) developed in INR Pitesti, and afterwards, RiskSpectrum Professional ([11], [12]). Reliability database for the TRIGA PSA model, initially being part of PSAMAN code was developed in RiskSpectrum Professional format, covering both reliability data for power and research reactors. Reliability data (failure rates, repair times, testing times) used in Level 1 and 2 PSA were chosen by appropriate criterion, priority having specific data of TRIGA reactor ([13]), then generic data taken from the other research reactors database.

3. Conclusions

The paper presents summary of PSA model for TRIGA 14 MW reactor which treated the following specific aspects:

- ✓ Initiating events;
- ✓ Core damage states;
- ✓ Radioactive release categories;
- ✓ Event trees and accident sequences;
- ✓ Reliability data.

Original results consist in qualitative and quantitative evaluation of event trees, accident sequences, fault trees. In the description of initiating events and event trees, deterministic analyses results for TRIGA reactor are referred and used.

The paper presents an example of initiating event, event tree for loss of flow accident (LOFA). Also, the headings which appear in the event tree for LOFA IE are mentioned and evaluated.

The reliability data used in the PSA model are based mainly on collection and processing of raw data taken from operation history of TRIGA reactor.

The PSA model uses the Romanian package code PSAMAN but also the well known swedish code RiskSpectrum.

The PSA model could help the operation of TRIGA reactor to improve the optimization of maintenance and test activity.

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