# THE LEVEL 1 PROBABILISTIC SAFETY ASSESSMENT APPLICATION OF TR-2


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ABSTRACT

This paper explains the outline of a level 1 probabilistic safety assessment (PSA) application for Turkish Research Reactor 2 (TR-2) using a plant specific reliability database.


## I. INTRODUCTION

In recent years, probabilistic safety assessment has acquired increasing importance. Most of the regulation authorities encourage greater use of this analysis technique to improve safety related decision-making and regulatory efficiency. The main objective of this study is to determine the strengths and the weaknesses of the TR-2 safety systems towards improving its design and operation.

This study is based on an independent, comprehensive and systematic safety evaluation of the Turkish Research Reactor-2 (TR-2). The probabilistic safety assessment (PSA) has been considered for full power state of the reactor. The TR-2 PSA represents a level 1 analysis, and was constructed using Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE). SAPHIRE is a PC-based general purpose tool for performing PSAs (INEEL). It is specifically designed to permit a listing of the potential accident sequences, compute their frequencies of occurrence, and assign each sequence to a consequence. SAPHIRE consists of several modules for fault tree analysis, event tree analysis, and for developing and maintaining component failure rate, and maintenance data [1].

The TR-2 PSA systematically identified and quantified the frequency of scenarios involving core damage by plant specific data. Scenarios resulting from both internal and external initiating events were analyzed. The resulting total core damage frequency of TR-2 was obtained.

## II. MAIN FEATURES OF TR-2

TR-2 is a 5MW swimming pool type research reactor that uses plate type fuel elements and is reflected by beryllium blocks. The reactor is moderated and cooled by light water.

The TR-2 reactor reached first criticality in December 1981, and it has been operated at its nominal power level of 5MW with few shutdown periods since October 1982. The TR-2 research reactor is being used for radioisotope production, for incore irradiation and training [2].

The cooling system of TR-2 is designed and equipped to satisfy 5MW thermal power level within a safe and continuous manner. This system is comprised of reactor pool, primary coolant loop, secondary coolant loop, cooling towers and auxiliary systems.

TR-2 is cooled by natural circulation up to a power level of 100kW and by forced downward circulation from 100kW to 5MW. The heat is dissipated from the primary loop through a plate type heat exchanger. The coolant is circulated by a pump in the secondary loop that traverses the secondary side of the heat exchanger and all of the six cooling towers.

Reactor control is provided by control rods. Scram is obtained by de-energizing the relays in the scram logic so as to disconnect them, and by cutting the voltages of control rod magnets so as to let them drop into the core by gravity.

Electrical energy for the reactor can be supplied by either one of the; gridlines, diesel generator unit, uninterruptible generator unit, uninterruptible power supply [2].

## III. OVERVIEW OF THE PSA MODEL

The PSA model was based on a large fault tree/small event tree (LFT/SET) approach. The SAPHIRE computer code was used to quantify the PSA model fault and event trees.

The IAEA technical document IAEA-TECDOC-400 [3] is taken as a reference to get a preliminary list of initiating events. Then, by using the results of the deterministic analysis that have been done before for TR-2, some of the initiating events are eliminated due to their very low occurrence rate and because they are out of the scope of level 1 probabilistic safety analysis. Excess reactivity insertion, loss of flow accident, flow blockage, loss of coolant accident, loss of off-site power, super prompt criticality internal, and earthquake external initiating events are determined for this analysis.

Plant design and operational data was assembled from existing safety analysis report of TR-2 [2].

Fault trees were developed for all the safety and safety related systems, and most of the support systems required for operation of safety systems.

In order to generate plant specific failure data, maintenance reports, and control room logs were examined in detail. DES data entry system [4] was used to input and record raw failure data. All of the related components were classified according to their category, group, and type. In compiling the data, the number of components, operating time, number of demands, failure mode, and number of failures were used to determine failure rates. Then, the compiled data is used as input for failure rates and uncertainty distributions of the basic events in the computer program. The error factors for the uncertainty distributions are derived from NUREG/CR-4550, Analysis of Core Damage Frequency from Internal Events, U.S., NRC, by [5]. Common cause failures were modeled using -factor model. The initiating event frequencies are obtained from plant specific, generic data and by estimations.

The study was developed for full power operating conditions of TR-2. The reactor core was the only source of radioactive hazard considered, and a "core damage" state was conservatively defined on the basis of the peak cladding temperature of 550$^{o}$C.

Success criteria were derived through plant specific thermal-hydraulics calculations from safety analysis report [2].

## IV. RESULTS OF THE PSA

The TR-2 PSA systematically identified and quantified the frequency of scenarios involving core damage. The frequencies of core damage caused by different initiating events are given in Table 1. The results indicate the 90% confidence range for the frequency of damage, as measured by the 5 $^{th}$ and 95 $^{th}$ percentiles of the uncertainty curve.

The mean, or average total core damage frequency is 1.05 x $10^{-3}$ events per reactor year. This is the frequency measure that is to be compared most meaningfully to the safety objectives. The contribution percentages of initiating events to the total core damage frequency are given in Table 2.

Core damage caused by loss of off-site power initiating event has the largest contribution. Then come the loss of flow, flow blockage, and earthquake events. Super prompt criticality, loss of coolant, and excess reactivity insertion events have little effects on core damage frequency.

## V. CONCLUSION

The results of this study indicate the mean core damage frequency of TR-2 for the stated initiating events to be 1.05E-3/RY. That is, core damage may occur once in 952 reactor years. The difference between the reactor calendar life and the cumulative number of operating years must be emphasized again. For the case under consideration, the calendar life of TR-2 is 13 years, with the actual period of operation being only 1.7 reactor years.

Most regulatory authorities do not use formal acceptance criteria with respect to the final numerical results of the PSA. Reasons are the absence of standardized PSA methods, the scatter and uncertainties in the input data. However, some regulatory authorities have established such criteria -not in the sense of mandatory safety limits, but rather as yardsticks supporting the evaluation of PSA- results. The probabilistic safety criteria of IAEA for power reactors level 1 result is assigned a frequency of one core damage event in 10,000 years of plant operation for existing plants.

The core damage frequency of TR-2 is approximately ten times larger than the acceptance criteria of IAEA for existing plants. As have been mentioned this is not however a mandatory safety limit. At the same time, it can be said that the PSA results of TR-2 are conservative. The assumptions made in generating failure data, such as the acceptance of a failure being about to occur in case the respective failure had not occurred up to the freeze date, is a conservative approach. Since the failure data for TR-2 in its early operating years is lacking, the plant specific data generated is mainly a product of last four or five years of operation. And the quality of this data needs assessment.

The limitations on probabilistic safety assessment of TR-2 can be qualified. These are due to the fact that the results of a PSA invariably contain uncertainties arising from three main sources:

- Uncertainties due to a lack of comprehensive data regarding the area under consideration. The internal and external fires, internal flooding, aircraft crash, and also a detailed seismic analysis must be considered in a complete PSA of TR-2.
- Uncertainties regarding data. Such uncertainties concern the reliability data for plant components, the frequency of initiating events, common-mode failures and failures resulting from human actions. The main uncertainties are those relating to the frequency of rare initiating events, as well as data relating to human factors.
- Uncertainties associated with modeling assumptions that cannot easily be quantified, such as the resistance of certain components under accident conditions, poorly understood physical phenomena or human actions.

Despite these uncertainties, the assessment of both the strengths and weaknesses of the safety features will clearly suggest ways of improving both the design and operation of TR-2. Probabilistic safety analysis has thus become an important supplement to deterministic analysis in checking the safety level of a facility and improving its reliability by identifying design weaknesses.

The PSA application results of TR-2 show that most of the core damage sequences accompany the cut sets of failure of UPS, and failure of the scram system. Concluding suggestions towards an acceptable PSA result for TR-2 would be primarily to improve the electrical system with an additional uninterruptible power supply, and secondarily to diversify or improve the shutdown system for both internal and external events.

REFERENCES


1. USNRC, *Systems Analysis Programs for Hands-On Integrated Reliability Evaluations (SAPHIRE) "The Basics"Manual,* Washington D.C., 1997.
2. ÇNAEM, *Safety Analysis Report of TR-2,* Istanbul, 1997.
3. IAEA, *Probabilistic Safety Assessment for Research Reactors,* IAEA-TECDOC-400, Vienna, 1986.
4. IAEA, *Generic Component Reliability Data for Research Reactor PSA*, IAEA-TECDOC-930, Vienna, 1997.
5. Modarres M., M. Kaminsky and V. Krivtsov, *Reliability Engineering and Risk Analysis,* Marcel Dekker, New York, 1999.
6. IAEA, *Working Material, Application of Probabilistic Safety Assessment to Research Reactors,* IAEA-TECDOC-517, Vienna, 1989.

**Table 1**. Core damage frequencies

| Category | Frequency (events per reactor year) | | |
|---|---|---|---|
| | **Mean** | **5th** | **95th** |
| Excess Reactivity Insertion | 9.18E-8 | 5.71E-8 | 9.58E-8 |
| Loss of Flow | 6.20E-5 | 5.26E-5 | 8.10E-5 |
| Flow Blockage | 6.63E-5 | 3.77E-5 | 1.03E-4 |
| Loss of Coolant | 1.56E-7 | 1.54E-7 | 1.59E-7 |
| Loss of Off-Site Power | 2.47E-3 | 7.54E-4 | 5.87E-3 |
| Super Prompt Criticality | 1.00E-5 | 1.00E-5 | 1.00E-5 |
| Earthquake | 4.82E-5 | 4.82E-5 | 4.82E-5 |
| Total Core Damage | 1.05E-3 | 9.03E-4 | 6.12E-3 |

**Table 2**. The importance order of accidents

| Category | Contribution Percent % |
|---|---|
| Loss of Off-Site Power | 83.23 |
| Loss of Flow | 5.94 |
| Flow Blockage | 5.36 |
| Earthquake | 4.50 |
| Super Prompt Critical | 0.95 |
| Loss of Coolant | 0.01 |
| Excess Reactivity Insertion | 0.01 |