PSA SUPPORT SAFETY ANALYSIS USING RELAP5 FOR THE REACTIVITY INSERTION EVENT AT TRIGA 14 MW REACTOR

Mirea MLADIN[1], Daniela MLADIN[1], Ilie PRISECARU[2], Nicolae DANILA[2], Daniel DUPLEAC[2]

The paper presents the deterministic support analysis in case of Reactivity Insertion Accident (RIA) considered as initiating event in the PSA project. It studies the reactivity worth necessary to damage the research reactor fuel. Previously in the PSA study the postulated initiating event of error in the fuel manipulation was considered as a result of dropping one or maximum two fuel bundles from the lifting device in case of core configuration rearrangements.

This type of event was actually produced in the nineties. The paper gives some elements of the previous PSA model with respect to this initiating event: the event tree and the results of the accident sequences. The focus of the paper is on the results of thermalhydraulic code RELAP5 Mod 3.2 which uses a point kinetics model for studying the transient in case of different reactivity worths and insertion times.

The results include evolutions of heat transfer mode, maximum temperature inside fuel elements and peak values of the power excursion. The conclusions highlight the possibility of infringement of the safety criteria for the TRIGA SSR 14 MW reactor during the analyzed transients and also discuss the necessity of including this event in the PSA model.

Keywords: reactivity insertion analysis.

1. Introduction

In 2005, in a previous stage of the PSA model development for TRIGA 14 MW, a series of initiating events (IEs) were defined and the corresponding event trees were constructed. Among the IEs, the human error in the fuel manipulation was considered as a result of dropping one or maximum two fuel bundles from the lifting device in case of core configuration rearrangements. This type of event can lead either to the mechanical damage of the fuel or to a power excursion that can thermally damage the reactor core. The power excursion as a result of such an event actually happened in the nineties. The outcome of such a power excursion depends on the existence of sufficient core subcriticality shutdown margin for

[1] Safety Analyst, Institute for Nuclear Research, Romania
[2] Prof., Department of Energy Production and Use, University "Politehnica" of Bucharest, Romania

limiting the worth of possible reactivity insertion. The Operating Limits and Conditions (OLC) stipulate that the reactor core has to be configured in such a way as to be subcritical by at least 0.55$, provided that the most reactive control rod is out of the core.

During a reevaluation process of the PSA model, in 2006, the possibility for a thermally induced damage of the core as a result of reactivity insertion was questioned and analyses were performed in order to verify the positive reactivity worth that would produce the infringement of the safety criteria. Also, to verify whether or not this worth can possibly be introduced into the core in a real case.

The Final Safety Analysis Report (FSAR) indicates the fuel centerline temperature as the safety limit parameter, both in forced cooling and in transients in which the clad can reach the same temperature as the fuel, and establishes threshold values for these situations. In case of a Reactivity Insertion Accident (RIA), the fuel temperature value of 940 ˚C is applicable, calculated in FSAR [1] on the basis of hydrogen release from the $U\text{-}Zr\text{-}H_{1.6}$ fuel matrix and subsequent pressure induced on clad material. But the transient Departure from Nucleate Boiling (DNB) condition can also be attained. The thermalhydraulic analysis for RIA aims at determining the power evolution and the heat removal yielding the clad and fuel temperatures for groups of fuel elements one of which is representing the maximum rated fuel pin in the reactor. For this purpose, a RELAP5 model for the reactor core and reactor pool was used, including reactor kinetics input description.

First, the paper will describe the event tree associated to the fuel bundle(s) manipulation error and then, it will focus on the analysis performed with RELAP5.

2. Event tree for fuel manipulation error

The event tree is given in Fig. 1. Once the fuel bundle has been dropped (or the operated fuel bundle and another bundle accidentally attached to it) the tree tests the functioning of the main pumps power supply (S1) in order to cool the core. The tree was constructed for the power excursion following the dropping, in other words a reinsertion of the bundle(s) in the core under the action of gravity. If mechanical damage occurs, there would be no need for primary cooling which would only spread in the facility the fission products.

Then, in the tree, follows successively the test of:

- the primary cooling system itself (PRIMARY);
- the electrical power supply of the emergency cooling pump (TCTAT1);
- the emergency cooling system (RACAVAR);
- the emergency ventilation (VENTDOZ), if the emergency power supply (TCTAT1) is available.

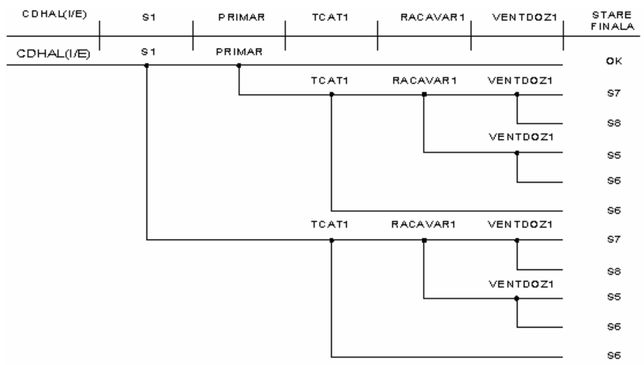

Fig. 1. The Event Tree in case of RIA due to fuel manipulation error.

If cooling is not available, several generic final states were defined in that stage of the project, due to the unknown core fraction if any that would be damaged by such a power excursion. The event tree and the associated fault trees were solved using the PSAMAN package and core damage frequencies (CDF) were obtained.

3. RELAP5 model for the reactivity insertion in TRIGA 14 MW

Fig. 2 presents the control volumes and connections between them in the RELAP5 [2] model for TRIGA reactor core and open pool. There are four hydrodynamic channels in the core: the hot channel, the average channel, the core by-pass channel (shroud clearances and water inside control rods) and experiments water channel.

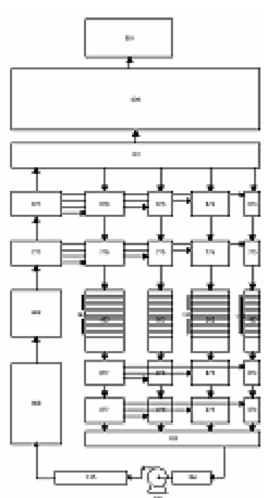

Fig. 2. Nodalization of the TRIGA reactor core and pool.

The model is representing the standard core configuration of the reactor with 29 Low Enriched Fuel (LEU) bundles, summing up to 725 fuel pins inside the core. Using this basic model, information concerning the reactor kinetics model in RELAP5 was supplied.

TRIGA reactors are considered inherently safe due to a large prompt negative reactivity coefficient. The phenomena that contribute to it are:

- cell-increased thermal disadvantage factor;
- irregularities in the fuel lattice due to control rods;
- Doppler broadening of the ^{238}U and Erbium resonances;
- leakage from the reactor core.

Cell calculations of the LEU negative coefficient were performed with WIMS/D4 code in [3]. From these was derived the dependence of the reactivity coefficient on the fuel temperature:

$$\alpha(T) = 0.0022 + 2.\times 10^{-6} * T - 6.\times 10^{-10} * T^2 \tag{1}$$

Using this formula (T in Rankine degrees), negative reactivity introduced by the fuel heating from temperature T_0 to the temperature T is given by:

$$r(T) = \int_{T_0}^{T} \alpha(T)dT \tag{2}$$

In formula (2), T_0 is given a value of 50 ˚C estimated to be the shutdown average fuel temperature.

In this way, the values for the fuel temperature feedback were calculated and are presented in Table 1.

Table 1

Feedback reactivity on fuel temperature (left) and on moderator density (right)

Temperature (K)	Negative reactivity ($)
423.15	0.588
523.15	1.212
623.15	1.864
723.15	2.538
823.15	3.225
923.15	3.920
1023.15	4.615
1123.15	5.303
1223.15	5.977
1323.15	6.631
1423.15	7.257
1523.15	7.848
1623.15	8.397

Moderator density (kg/m^3)	Negative reactivity ($)
9.98e-3	14.70
99.84	13.30
199.68	11.66
299.52	9.98
399.36	8.32
499.20	6.71
599.04	5.17
698.88	3.72
798.72	2.38
898.56	1.14
990.00	0.00

Moderator feedback contribution was obtained by calculating the neutron multiplication factor with WIMS [4] code in infinite lattice assumption and varying the atom densities of the hydrogen and oxygen inside the coolant region of the neutronic cell. The density dependence of this contribution is also given in Table 1.

4. Reactivity insertion transients

A series of calculations were performed for the insertion of different positive reactivity worths and insertion times. In all calculations, it was considered a linear insertion on the time interval, from the negative shutdown reactivity (chosen as -7.55 $) up to the positive worth.

First, an insertion of 1$ worth was studied for 0.1 s, 1 s and 6 s insertion time. The maximum fuel temperature, the clad temperature, the reactor power and the heat transfer mode are presented for the 1$ in 1 s case in Fig. 4, 5, 6 and 7 respectively.

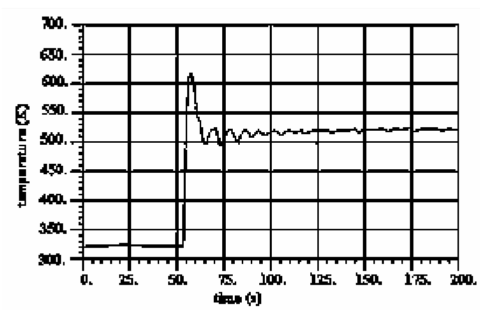

Fig. 4. Max. fuel temperature for 1$ in 1 s.

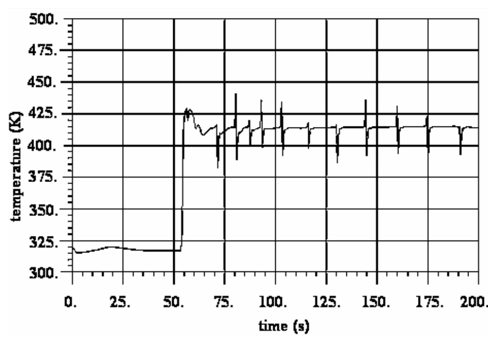

Fig. 5. Clad temperature for 1$ in 1 s.

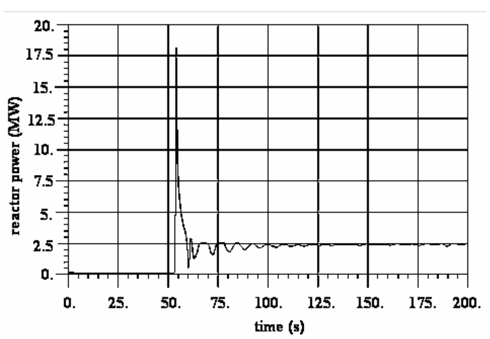

Fig. 6. Reactor power evolution for 1$ in 1 s.

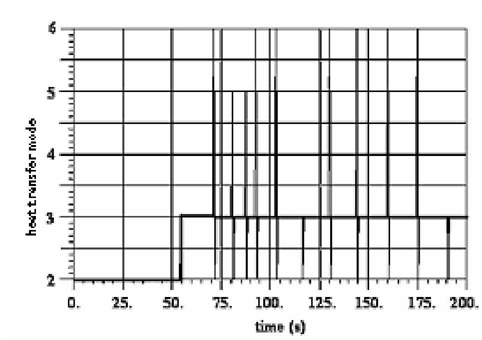

Fig. 7. Heat transfer mode for 1$ in 1 s.

There is a power peak of 17.5 MW (the power before transient is 0.081E+4 MW) and, in 5 seconds the reactor is stabilizing at about 2.5 MW through a series of oscillations. The power remains dynamically equilibrated by the reactivity feedback since all the control rods are fully in and no other control action was considered (shim bundle insertion or fuel removal from the reactor).

The maximum temperature in the fuel (power peaking for the maximum rated fuel element is 1.93) reaches 613 K (340 ˚C), and the peak cladding temperature is around 425 K (152 ˚C). The heat transfer mode in the hot channel after the insertion is basically nucleate boiling (htmode=3) with only very short excursions to transition boiling, during the oscillations. The transient does not produce damaging consequences for the reactor. Changing the insertion time to 0.1 s or to 6 seconds does not make a significant difference to the case analyzed (1$ in 1 s).

Reactivity insertion worth was progressively increased in the input deck. Fig.8 and 9 presents fuel and, respectively, the cladding temperatures for a 2$ insertion in 1 second. Fig.10 and 11 presents the evolution of the same parameters in case of a 2.4 $ worth in 1 s.

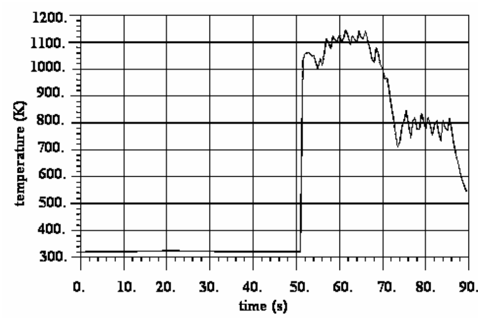

Fig. 8. Fuel temperature for 2$ in 1 s.

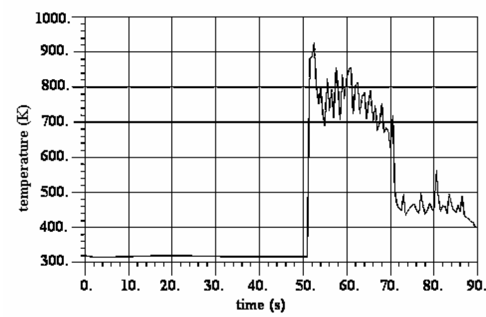

Fig. 9. Clad temperature for 2$ in 1 s.

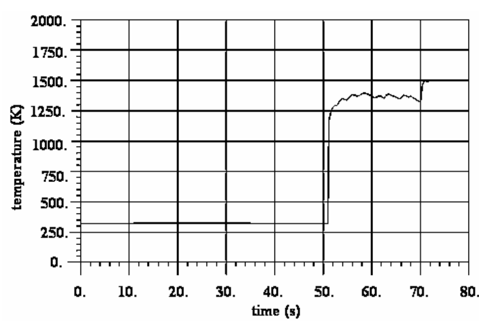

Fig. 10. Fuel temperature for 2.4$ in 1 s.

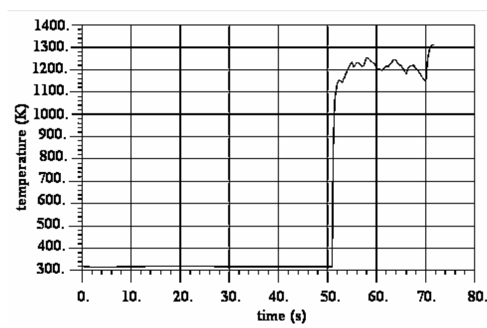

Fig. 11. Clad temperature for 2.4$ in 1 s.

The transients in Figures 8, 9, 10 and 11 are not extending up to 200 s as the input time limit was set, because of the numerical difficulties encountered by the code. Nevertheless, from these figures it can be seen that the reactor can still cope with the 2$ in 1 s insertion as the temperature in the fuel are below 940 ˚C which is the safety limit, but not with a 2.4 $ in 1 sec. The later case produces temperatures that are exceeding by far the safety limit (1500 K at 70 seconds and still increasing in the end), and clad rupture is expected to appear, at least for the maximum rated fuel pins.

Figures 12 and 13 present the reactor power evolution and the heat transfer mode for the 2.4$ insertion. There are power oscillations produced by the two opposing tendencies: the strong neutron multiplication factor of the core and the feedback reactivity. The former, producing the power increase, seems to predominate because the power peaks are higher towards the end of this calculation (73 s). The film boiling regime (htmode=7) is reported by the code from the beginning of the transient.

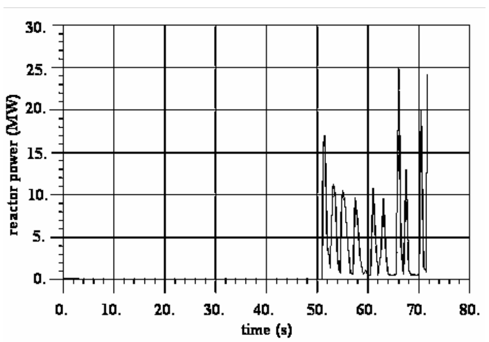

Fig. 12. Reactor power for 2.4$ in 1 s.

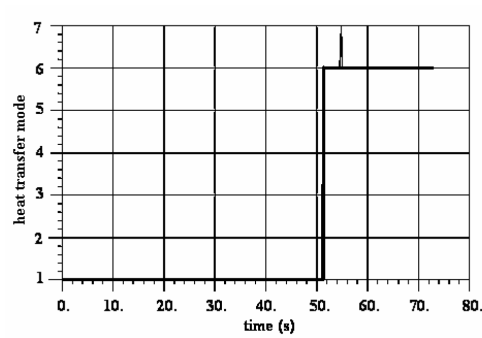

Fig. 13. Heat transfer mode for 2.4$ in 1 s.

Changing the insertion time to 0.1 s and to 6 s for the same 2.4$ insertion worth gives the maximum fuel temperatures evolution in Fig. 14 and Fig. 15. These transients are also leading to unacceptable temperature in the fuel and cladding, the faster insertion 0.1 s making the evolution of the temperature faster.

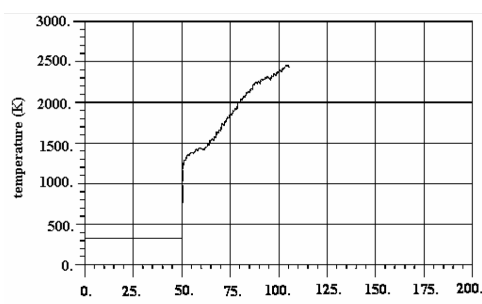

Fig. 14. Fuel temperature for 2.4$ in 0.1 s.

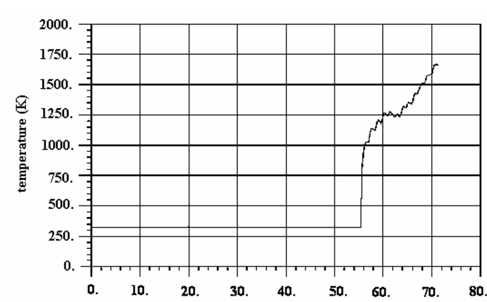

Fig. 15. Fuel temperature for 2.4$ in 6 s.

6. Conclusions

The above presented results indicate a necessary of 2.4$ positive insertion in order to damage the reactor as a result of the RIA power excursion.

In the calculations, reactivity insertions were introduced regardless the real possibility in the facility for such insertions, in order to test how much reactivity worth is needed to damage the reactor. In reality, the maximum conceivable reactivity insertion due to the reinsertion of two central bundles in the standard core configuration can be deduced from the [1] (page 2-138) where calculation results are provided for the replacement of two central fuel bundles by stainless steel shim bundles. The maximum value in such case is 5.43$. That would clearly be a conservative value because the shims themselves are absorbers and would make their contribution to this worth. Because of that, margin allowed for two bundles (more reactive) in another configuration. At the same time, the worth of the bank of rods in the shutdown state for the standard configuration is 19$, according to [1] (page 2-132). Even in the worst case, for a new configuration, in which only the rod stuck-out criterion from the OLC is fulfilled (0.55$), the reactor in the shutdown state will be subcritical by more than 7-8$. This is due to the fact that, in such a situation, a very reactive group of fuel bundles at the core centre should exist and the central rod will exceed the contribution to the bank in the standard core configuration (7.28$) . All rods are considered fully in for a secure shutdown state.

As a main conclusion of this, there is no conceivable way to compensate the bank worth and to produce an additional 2.4$ positive insertion with the reactor in the shutdown state, by accidental fuel insertion. This event will produce only a subcritical increase in multiplication of the reactor. Consequently, the initiating event due to fuel manipulation error was discarded because it will not produce fuel damage as a result of insertion of two bundles into the core. The only way to damage the fuel by a RIA transient would be to rapidly eject the control rods out of the core.

REFERENCES

[1]. *General Atomic Co*, Final Safety Analysis Report for Romanian TRIGA 14 MW, 1978.
[2]. *Idaho National Engineering Laboratory*, RELAP5/mod3 Code Manual, 1995.
[3]. *Gh. Negut, M. Mladin, I. Prisecaru, N. Danila*, "Fuel behavior comparison for a research reactor", E-MRS Spring 2005, May 31-June 3, Strasbourg, published in Journal of Nuclear Materials, **vol. 352**, 2006, pp. 157-164.
[4]. *K. Kowalska,* The S-WIMS Code for the CYBER-72 Computer, Institute of Nuclear Research, Warsaw, 1975.