SOME EXAMPLES OF ACCIDENT ANALYSES FOR RB REACTOR

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ABSTRACT

The RB reactor is heavy water critical assembly operated in the VINČA Institute of Nuclear Sciences, Belgrade, Yugoslavia, since April 1959. The first Safety Analysis Report of the RB critical assembly was prepared in 1961/62. But, the first accidental analysis was done in late 1958 in aim the examine power transient and total equivalent doses received by the staff during the reactivity accident occurred on October 15, 1958. Since 1960, the RB reactor is modified few times. Beside initial natural uranium metal fuel rods, new fuel (TVR-S types) from 2% enriched metal uranium and 80% enriched UO_2 were available since 1962 and 1976, respectively. Also, modifications in control and safety systems of the reactor were done occasionally. Special reactor cores were created using all three types of fuel elements, among them, the coupled fast-thermal ones. Nuclear Safety Committee of the VINČA Institute, an independent regulatory body approved for usage all these modifications of the RB reactor. For those decisions of the Committee, the Preliminary Safety Analysis Reports were prepared that, beside proposed technical modifications and new regulation rules had included analyses of various possible accidents. Special attention is given and new methodology was proposed for thoroughly analyses of design based accidents related to coupled fast-thermal cores, that include reactor central zones filled by fuel elements without moderator. In these accidents, during assumed flooding of the fast zone by moderator, a very high reactivity could be inserted in the system with very high reactivity rate. It was necessary to provide that the safety system of the reactor had fast response to that accident and had enough high (negative) reactivity to shut down the reactor timely. In this paper, a brief overview of some accidents, methodology and computation tools used for the accident analyses at RB reactor are given.

1. INTRODUCTION

The 'RB' reactor [1] is an unshielded critical assembly designed in 1958 to operate using heavy water and natural (metal) uranium fuel rods. In 1962, 2% enriched metal uranium fuel TVR-S type of ex-USSR origin became available and the first safety analysis report was written. A study of the RB reactor as possible source of fast neutrons began in 1976, when the 80% enriched UO_2 (dispersed in Al matrix) fuel of TVR-S type was bought in former USSR, too. These special RB reactor cores are described elsewhere [2 - 4]. Among them, more complex fast neutron fields, i.e., the internal neutron converters - INCs (1983 and 1998), and the coupled fast-thermal core - HERBE (1990) are designed in latter years. Simultaneously, Operational and Regulation rules and the Safety Analysis Reports were updated.

In the most of about 4000 experiments carried out up to nowadays, RB critical assembly operated as pure thermal heavy water reactor at power levels from 10 mW to 50 W. Initial core (in 1958) was loaded with natural (metal) uranium fuel. Power excursion accident, in which six personnel were heavy irradiated (one died), occurred at the reactor only six month after the first start-up. The initial accident analysis was done by simple assumption of power excursion according to exponential law with fixed reactor period. The accident analysis of the RB reactor is done in 1962 by using simple space-independent codes ran on analogue [5] or digital computers [6]. More lately – in beginning of nineties, that accident was analysed in more details [7] using more appropriate numerical codes. Basic description of that accident and results of power excursion analyses are given in section 2 of the paper

From 1962, safety assessment of the reactor operation with different cores became regular practice – the first Final Safety Analysis Report and Regulation Rules were written in 1962 when reactor control, safety and dosimetry systems were modernised to allow operation with 2% enriched metal uranium fuel elements of TVR-S type. Further refurbishments of the control equipment were done in 1982 (new start-up channels) and in 1987 when new neutron control and gamma dosimetry logarithmic channels were added in the control panel. Regulation and operation rules were updated accordingly. Modifications, mentioned above, converted the RB critical assembly to a flexible experimental reactor with 1 W nominal power. It operates usually from 10 mW to 50 W, and, in especial occasions, at 'very high power' up to 10 kW.

Among many different thermal cores, special attention requires core no 5/1973 designed in 1973 for irradiation purpose by neutrons outside of the reactor tank. Such RB reactor core, with central heavy water reflector and fuel elements distributed along the core peripheral, was used in the IAEA Project *International Inter-comparison of Neutron Accidental Dosimeters*. Reactor control system was modified to allow operation at power level of few kW. Experiment went without problems, but a recent analysis [8] has shown that reactor was operated at approximately 2.5 time higher power than declared by operating staff. It was consequence of fact that reading instruments for fission power were not calibrated for the new core configuration and safety analyses and operation rules were not carried out in full details.

When 80% enriched uranium fuel became available, initial studies of design of fast neutron fields behind or inside reactor tank started. Various 'experimental cores' were designed up to now. The safety analysis report for operation of the RB thermal cores with HEU was updated in 1977/1978 when new start-up channels are built-in as well.

The INC is thermal-to-fast neutron converter designed inside the RB tank. The fast zone (without moderator) of INC-1 was designed as an 80% enriched fuel elements annulus surrounded with blanket made of two layers of the natural uranium fuel elements in separate aluminium tanks. A central air hole is designed for irradiation purpose. Three different versions of the INC are developed (INC-1, INC-2 [2] and INC-3 [4]). The INC thermal zone is the RB thermal core of 2% and 80% enriched fuel elements placed in square lattice pitch of 12 cm surrounded with heavy water reflector. The recent INC-3 is designed with 80% enriched U fuel ring and thin Cd layer in the fast zone offering large space for irradiation by fast neutrons.

The HERBE [3] is coupled fast-thermal neutron core designed in the RB reactor for fast neutron flux intensity increasing in a vertical experimental channel placed in the centre of the fast core. Characteristics of the HERBE are determined by computer codes and in the experiments for verification. Description of the HERBE system is given in section 3 of the paper. New Final Safety Analysis Report (FSAR) and appropriate changes in Operation and Regulation Rules for the HERBE system are written in 1991. The FSAR of the HERBE includes thorough accidental analysis. Basic description of few possible accidents and analyses of results of power excursion analyses in the HERBE system are given in sections 3 and 4 of the paper. Analyses of the most dangerous accident of the HERBE system – flooding of the fast zone by moderator is given in more details in section 5, accompanied by new measurements of reactivity-time function of the RB reactor safety system.

2. ANALYSIS OF RB REACTOR 1958 ACCIDENT

2.1. Accident short description

Accident occurred on October 15, 1958, during an experiment carried out at the RB critical assembly (Figure 1). Personnel operated the reactor were in the reactor hall (Figure 2).

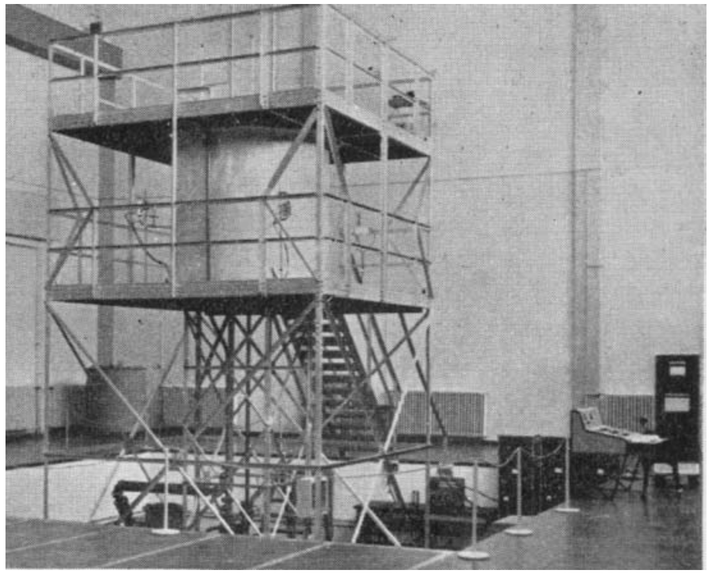

Figure 1 RB reactor in 1958

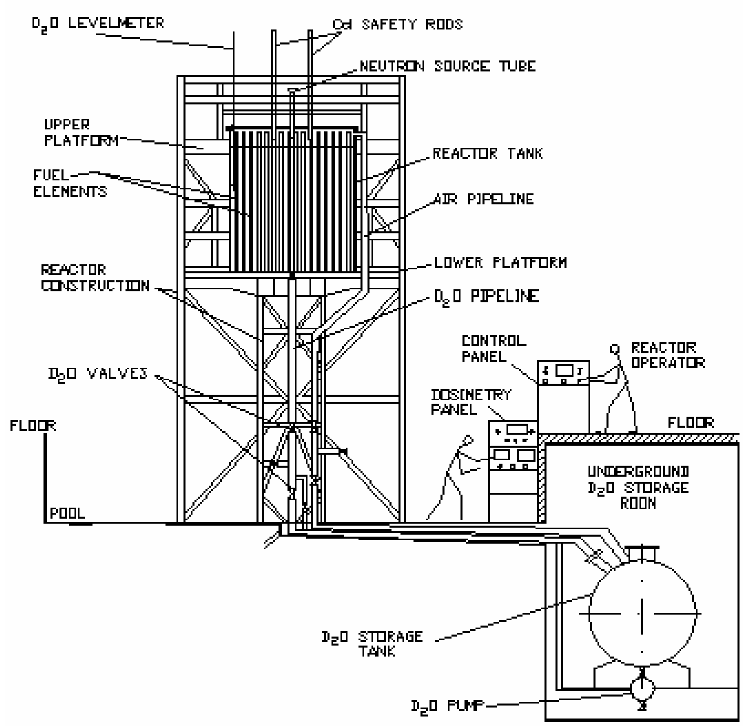

Figure 2 Scketch of RB staff position during 1958 accident

At heavy water level of 175 cm (3.5 cm below the expected critical level) a fast increasing of the moderator level (2.5 cm/min) was switched on (t = 0) in aim to achieve new (expected) subcritical level of 177 cm. During that process the staff was distressed by entrance of a non-staff person in the reactor hall. The D_2O moderator reached 177 cm level in the reactor core and continued to increase, because the operator did not switch off the pump. The instrumentation of the RB reactor used in dosimetry, alarm and safety systems was switched off or partially removed. After 84 s, the critical level (178.5 cm) was reached and the reactor has become supercritical. The reactivity and reactor power were continued to increase without any supervision of the staff and the whole amount of the heavy water was transferred from a storage tank into the reactor tank. The heavy water excess was 4.5 cm above critical level. Time duration of the power excursion was not recorded. Two BF_3 counters used by experimentators in the reactor hall, believed to work properly, had reached saturation level and were reading a constant maximum value even though the power was rising steadily. The third counter, behaved erratically, was disconnected. An automatic recorder for measuring airborne activity and radioactive fallout, 540 m away from the reactor building, registered the power rise and accompanying increased gamma background for an interval of approximately 10 minutes. The assembly was at this heavy water level for 433 s, when the staff personnel smelled ozone due to ionised air in the hall and realised that the system was supercritical. One staff member shut down the assembly manually using by two cadmium rods.Six staff personnel are exposed to high levels of radiation from neutrons and gamma rays. Estimated equivalent doses, received by personnel were about level of 50% of the lethal dose. One member of the staff died few days later in spite medical treatment that included bone marrow transplantation, as well.

2.2. Accident analyses

The first analysis of the RB 1958 accident, done in 1958 and 1960, was based on a simple approximation of the power excursion by an exponential function with a 10 s period. According to measured activity of irradiated Au and Cu foils found in the RB building and metal objects carried by irradiated employees, it was estimated [9] that total fission energy generated in the accident was 80 MJ ($\sim 2.6 \cdot 10^{18}$ fission). Recent calculations [7] using MACAN [10] and SCM [11] codes have shown that the RB reactor at D_2O level of 175 cm was subcritical with reactivity of $-0.305\ \beta_{eff}$. For heavy water excess of 4.5 cm, the system was supercritical with reactivity of 0.375 βeff. The power proceeded to rise, with a period of 12.3 s, to maximum level of nearly 2.5 MW, with the total released energy of 80 MJ (Figure 3).

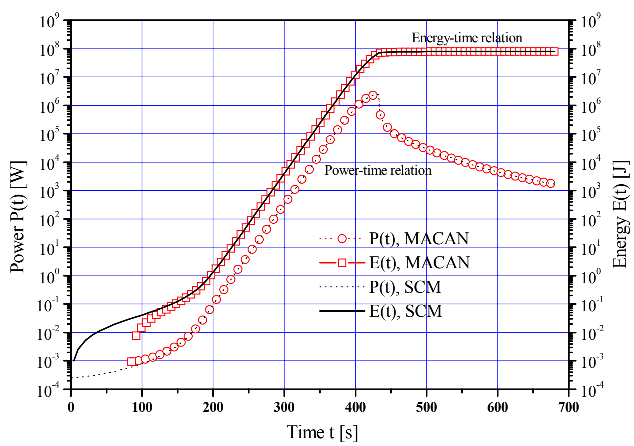

Figure 3 Power and energy vs time in RB 1958 accident

3. SAFETY ASSESSMENT OF RB IN-CORE FAST NEUTRON FIELDS

Development of all versions of fast neutron fields at the RB reactor was reviewed and approved by the Nuclear Safety Committee, an independent expert body of the Vinča Institute. The PSARs, FSARs and Operation Rules are reviewed and approved by the Nuclear Safety Committee, and proposed to the Director General to issue permission for regular operation of the reactor. General demands, set during design of these fast neutron fields at the RB reactor, were:

(a) Modifications of the RB reactor core should not be large;

(b) Only existing nuclear fuel elements can be used;

(c) The whole coupling (RB reactor thermal core - fast neutron field) should be strong in such way that coupled system acts a common thermal reactor (large prompt neutron lifetime) so that the existing RB reactor control system can operate normally; and

(d) The coupling should be designed in such way that the coupled system can be shut down quickly and safely with safety rods in thermal core.

All demands that were set for design of the fast neutron fields at the RB reactor were achieved. The safety analyses showed that the RB reactor operation with designed fast neutron fields is safe, without any need for significant modification of control or safety systems. The existing safety rods system has enough reactivity that can be inserted in a very short interval so any power excursion can be stop. Response times of the reactor control instrumentation are so that all control of the designed couplings is within normal operation modes of the RB reactor. According to the results of these analyses, the safety system of the RB reactor can quickly and safely shutdown the reactor during the most probable accident or in case of accident in which highest reactivity is inserted. Neither the system components nor the reactor stuff should be exposed to high doses (up to 25 mGy) during these postulated serious accidents. In aim to increase sensitivity of the safety system to the most dangerous possible accident (flooding of the fast zone of the INCs and HERBE by heavy water from thermal core), two moderator leak sensors (DCM) are placed in the outermost tank of the fast core and independently connected to the existing RB reactor safety system.

Presently, RB reactor operates with the core designed as the third version of the INC, INC-3 [4]. This special core is used for irradiation purposes, development of modern radiation protection, reactor control and safety systems, and for verification of new computer codes used for reactor design and safety studies developed at NET Laboratory in the Vinča Institute. Another field of the RB reactor recent extensive application is increasing interest for compilation and systematisation of evaluated benchmark experiments in critical safety. Three separate evaluations of the more than 20 carefully selected, well documented and reviewed RB reactor experiments, are included into the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook, managed by the OECD/NEA [12] and issued each year as CD ROM edition or Web presentation.

3.1 Description of the HERBE coupled fast-thermal system

Fast region of the HERBE is formed as three-zone system in the centre of the RB reactor. The central zone is 'fast core' (FC) designed from natural uranium fuel elements in the first aluminium tank (200/202 mm diameter) with axial vertical experimental channel (VCH). The fast core is surrounded by 'neutron filter zone' (NFZ) designed from Cd foil (1.6 mm thick) and natural uranium fuel elements in second aluminium tank (300/302 mm diameter). 'Neutron converter zone' (NCZ), which surrounds the FC and NFZ, is formed from 80% enriched UO_2 fuel elements in third aluminium tank (400/408 mm diameter). Each aluminium tank is closed at the bottom and waterproof so there is no moderator in the fast zone. Total height of the fast zone is 139 cm. Thermal core (driver) is designed from 44 fuel elements with 80% enriched UO_2 placed in 12 cm square lattice pitch of heavy water. A coupling zone between fast and thermal region is formed from 7 cm thick heavy water. Reflector of heavy water surrounds thermal core in the RB reactor tank (100/101 cm diameter). Horizontal cross-section of the HERBE coupled fast-thermal system and the enriched uranium TVR-S fuel element is shown in Figure 4.

3.2 RB reactor safety system basic characteristics

The reactor power is monitored by gamma compensated neutron sensitive ionisation chambers from very low power level (10 mW) corresponding to the direct current values of 1.0 pA - 10.0 pA, depending on particular neutron chambers sensitivity. The response time of reactor power instrumentation channels at so low currents (at the reactor criticality) is relatively long (1.5 s for linear DC power channels, 1.0 s for logarithmic DC power channel, and 4.0 s for associated periodmeters). The RB reactor safety system is based on the safety chain designed at safety logic 'one of two'. In such way, RB reactor operation is stopped by triggering any of 18 safety thresholds (SH) built in the safety system:
(a) AC electric power failure (3 SH) and DC electric power failure trip (1 SH);
(b) Linear power channels overpower trip (3 SH);
(c) Logarithmic power channel overpower (3 SH) or minimum period (3 SH) trips;
(d) Dosimeter channel gamma-ray overdose rate trip (1 SH);
(e) Linear and logarithmic power recorders overpower trip (2 SH),
(f) Moderator leaking sensors (DCM) in the HERBE fast zone trip (2 SH).

As result of the safety system activation, all four safety rods drop into the reactor core and pumping of heavy water into the reactor tank is stopped.

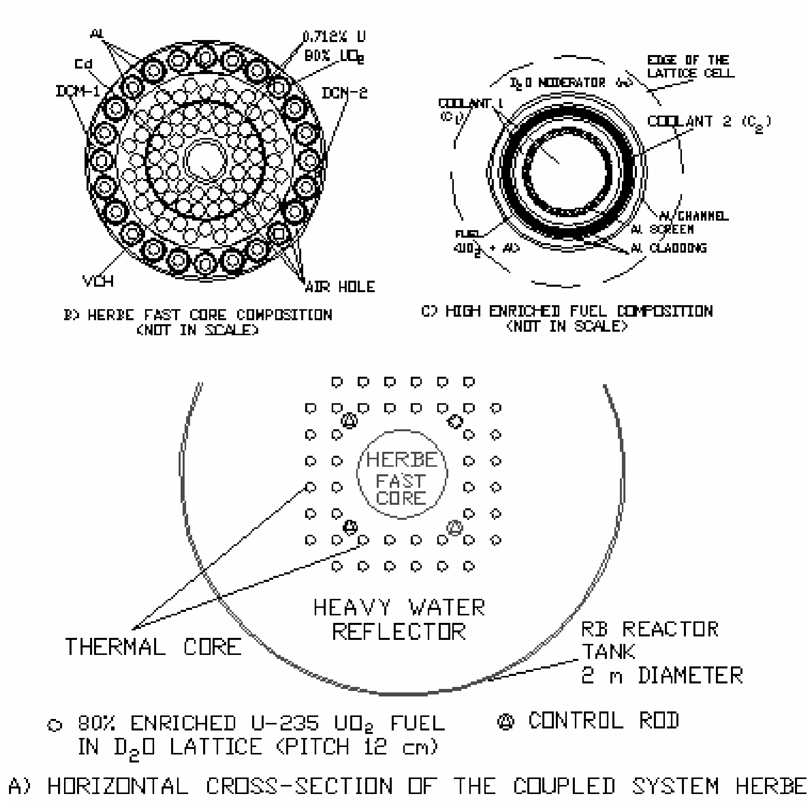

Figure 4. Horizontal cross-section of the RB coupled fast-thermal system HERBE

4. ANALYSES OF THE HERBE ACCIDENTS

Majority of safety analyses of the HERBE system are based on the RB reactor previously safety analysis reports and more than 30 years of operational and safety experience. The analyses are presented in full details in the HERBE FSAR [13]. As possible main causes of accidents during operation of the HERBE coupled fast-thermal system are accepted and were analysed following actions: (a) Increasing heavy water level over critical level; (b) Control (safety rod # 3) rod withdrawn at criticality; (c) Suddenly filling of an experimental channel in the thermal core with moderator; (d) Moving out an experimental or other part of equipment from thermal core, and (e) Suddenly flooding of the fast zone with moderator from thermal core.

4.1. Increasing heavy water over critical level

The criticality of the RB reactor is attained and all power changes are performed manually by operator action, changing heavy water level in small increments or decrements. Thus, increasing of heavy water level over the critical level is recognised as the most probably accident due to the operator error during operation of the RB reactor. Interlock system is designed in such way that moderator increasing is turned off automatically after each 60 s, resulting in heavy water level increase for only 8 mm. Heavy water gradient at critical level was measured as $(191.6 \pm 1.5)\cdot10^{-5}$ cm^{-1} which is very near to the calculated one $195\cdot10^{-5}$ cm^{-1}. This increase of heavy water can be represent as a reactivity-time ramp function with reactivity rate of $2.6\cdot10^{-5}$ s^{-1}, which is very slow and easily controllable by the operator himself or the safety system of the RB reactor. Only in the case of partially failure of the RB safety system and great carelessness of the reactor staff in the control room, this slow increase of heavy water can drive reactor power to high level.

4.2 Withdrawn of safety rod at critical level

The RB reactor interlock system is designed in such way that simultaneously withdrawn of two safety rods is not possible. In case of the HERBE system the low reactivity control rod is replaced by new designed high reactivity safety rod SR#3. The SR#3 is the last withdrawn from the reactor core at subcriticality of $k_{eff} \sim 0.8$. This standard operation has possibility to result in accidental situation only in the case if the actual value of the k_{eff} is (by mistakenly judged to be lower) very close to 1.0, which can be recognised easily by the RB reactor power channels (of the safety system) and the staff in the control room.

4.3. Filling of an experimental channel with moderator

The RB reactor is designed in such way that different experimental channels can be placed in the core. Suddenly rapture in a vertical experimental channel (with maximum possible diameter, placed at the highest reactivity position in the thermal core) can be represent as a ramp reactivity-time function with reactivity rate of $31\cdot10^{-5}$ s^{-1} in the first 10 s, during which the heavy water moderator floods the experimental channel completely. It was shown that the safety system could safely stop the reactor power excursion.

4.4 Moving out an experimental equipment from the reactor core

According to the RB reactor Operation rules, the experimental equipment (detectors, samples, etc.) can be placed in the reactor core only if its reactivity is less than $200\cdot10^{-5}$.

Moving out this equipment during the RB reactor operation at the critical level (at approximately 10 mW) can be performed only under supervision of the reactor staff. It was shown that the safety system could safely stop the reactor in case of this accident.

5. FLOODING OF HERBE FAST ZONE WITH MODERATOR

It was accepted that there is a finite probability that the external Al tank (4 mm thick) of the HERBE fast zone can be suddenly broken at welding position in 1 mm width around the whole tank circumference, at height of 1 m, or at the bottom of the tank. This situation enables a penetration of moderator from the thermal core into the HERBE fast zone and results in high and fast reactivity increase of the entire coupled fast-thermal system. For that reason, the maximum of the safety philosophy is applied in construction of the HERBE:

(a) Three fast zone separate Al tanks, each closed at the bottom, are designed and were checked for water leaking and welding quality;
(b) Enriched fuel elements in the NCZ are placed in sealed Al channels so that moderator could not enter in;
(c) Fuel segments in each fuel channel in the NCZ, are replaced, at the channel bottom, by Al supporters (43.0 cm long) closed in such way that the heavy water could not penetrate in;
(d) Two separate moderator-leaking detectors (DCM) are placed into the NCZ and connected at different places in the reactor safety system;
(e) Low reactivity control rod is replaced by new designed high value reactivity rod SR#3.

It was assumed that breaking more than one Al tank in the same time (of existing three ones forming the HERBE fast zone) had not high probability. Determinations of true reactivity-time dependence during flooding of the NCZ and timely action of the HERBE safety system in the accidental analyses were of the major importance. Specific safety experiments are carried out at RB reactor with HERBE core in aim to determine reactivity of the reactor safety rods, and their drop in times. Additionally, the special experiment with controlled flooding of the NCZ by moderator is done in aim to verify calculation results.

5.1. Determination of the HERBE safety rods reactivity-time function

HERBE system is designed with 4 safety rods. Two rods are the “regular” safety rods (SR#1 and SR#2), which, after the safety system was activated, drop into the core in short time without any pause during movement. Third safety rod (SR#3) is designed instead control rod and after triggering of the safety system drops into the core with one cessation during movement. The heavy water level meter (WLM = PN) acts in the safety system as fourth safety rod (SR#4) that drops into the core with two cessations during movement.

The accurate determination of the rod drop times is requested by the Nuclear Safety Committee and was performed in the specific experiments in the HERBE system. In the initial experiments, an electromagnetic transducer of rod's motion into voltage and *V-t* conversion by ADC in a computer [14] were applied. Evaluation of the experimental data, under approximation of constant acceleration during motion and instantaneously interruption of motion, has shown a very complex timing of the HERBE safety rods. In the later phase, a digital optical incremental device was used [15] for on-line data measurement of functions of position-time for safety rods SR#3 and SR#4 that have cessations during drop time. Results of measured trajectories are shown in Figures 5 and 6, respectively. Averaged measured trajectories (“motion law”) for all four rods of the RB reactor safety system are shown in Figure 7.

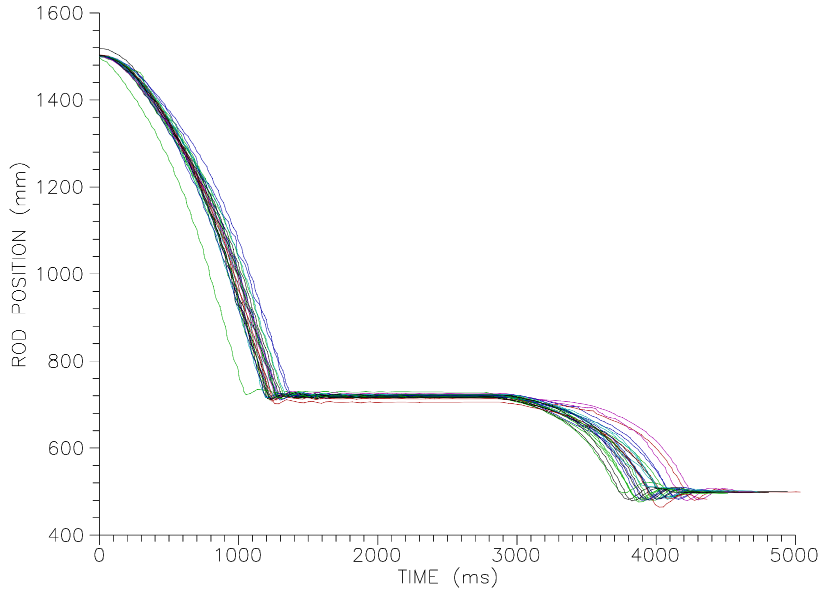

Figure 5. Measured trajectories of SR#3

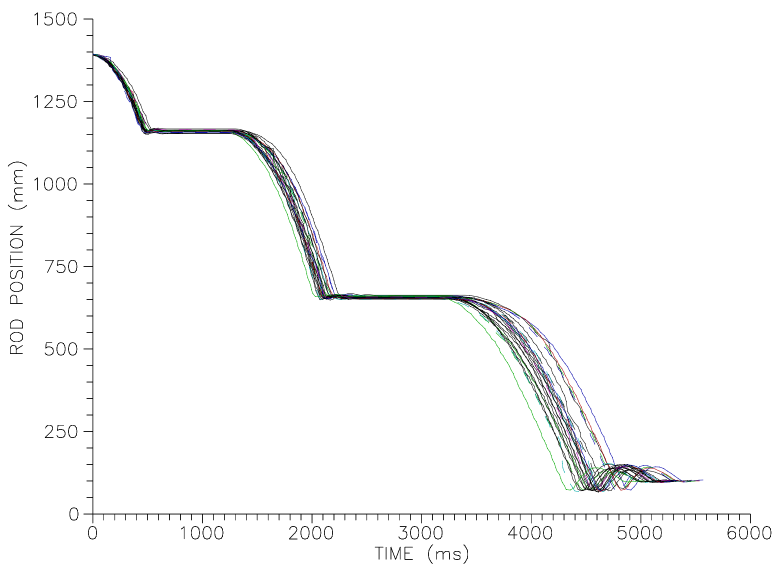

Figure 6. Measured trajectories of SR#4

Reactivity worth of safety rods of HERBE was determined in series of measurements and compared to calculated ones. In the experiments, each safety rod was inserted in the core separately or in combination with the others. Neutron time distribution after shutdown was recorded using a BF_3 counter. After data smoothing reactivity was determined from the known neutron flux-time distribution using IM code [17] based on the inverse kinetic method. The HERBE system kinetic parameters (β_{eff} and Λ) were determined using computer code AVERY [18], and verified in separate experiments [19]. The normalised reactivity of the water level meter (WLM), as a function of normalised rod position in the HERBE system, is approximated by polynomial function of 5th order determined by the best fit of experimental data. It was used in the final determination of complex safety rods reactivity-time function shown in Figure 8.

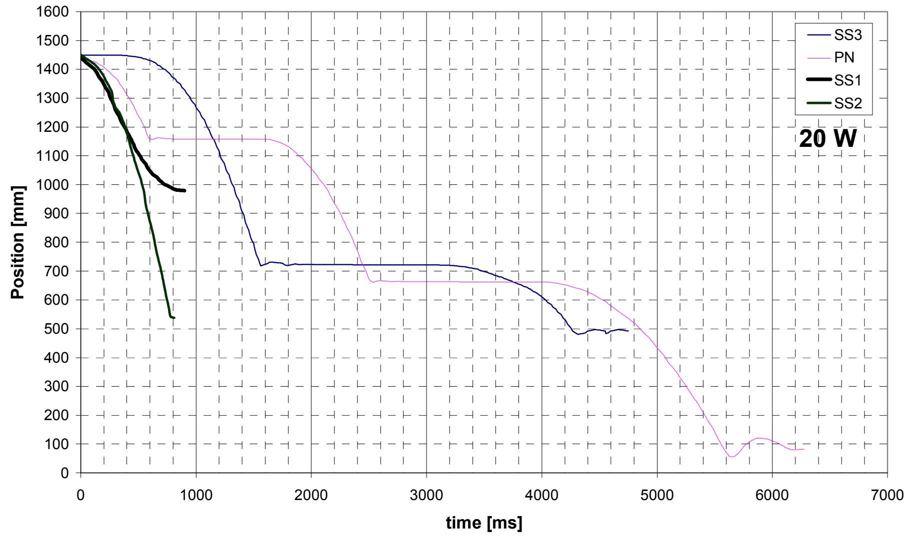

Figure 7. Measured motion laws for safety rods of HERBE system

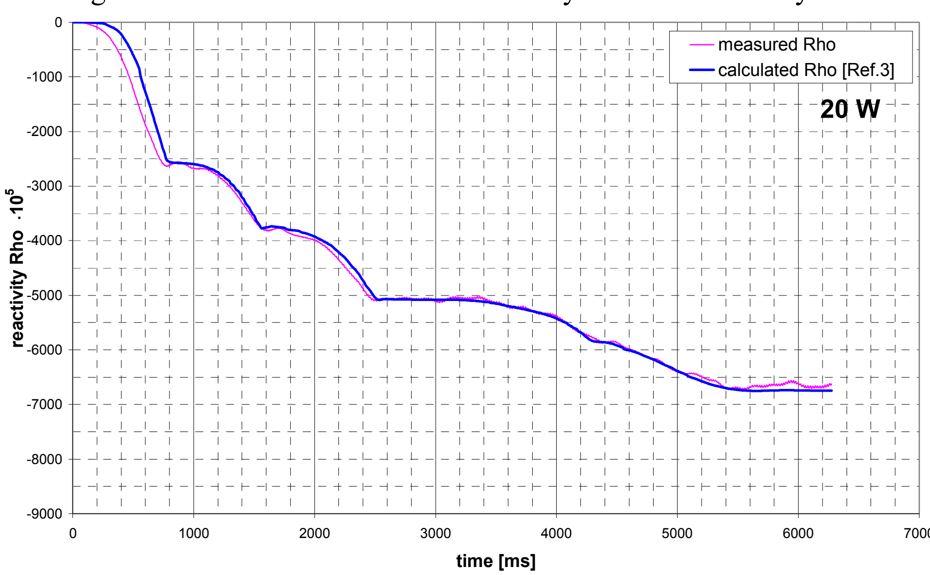

Figure 8 Reactivity-time function for HERBE safety rods

5.2 External Reactivity-time function during NCZ flooding

All possible construction activities were done to prevent heavy water from entering in the HERBE fast zone in a case of accidental flooding. These construction details reduced the total reactivity excess for almost 50 % and reduced reactivity insertion rate to acceptable value controllable by the safety system of the RB reactor. In aim to verify calculation result for this reactivity value, the special experiment with controlled flooding of the NCZ is performed. The decanting device is designed for semiautomatic transfer of the moderator from the HERBE thermal core into the NCZ. Due to safety precautions, this operation is performed at the reactor being high sub-critical. For each heavy water level in the NCZ the critical level of the system is measured by standard criticality approach procedure. Results of the experiments and calculations [3] were shown acceptable agreement, confirmed also by recent calculations done by the MCNP code. A simple flooding model was applied [20] and gave 13-16 s filling time of the NCZ with the moderator after assumed instantaneous break of 4 mm thick external Al tank in 1 mm width through whole circumference. Increase of the heavy water level in the NCZ during flooding time is used to calculate change of the reactivity with increasing of flooding moderator height in the NCZ, determined by computer codes. This “external” reactivity-time function, during heavy water flooding time of the NCZ, calculated and experimentally verified, is shown in Figure 9.

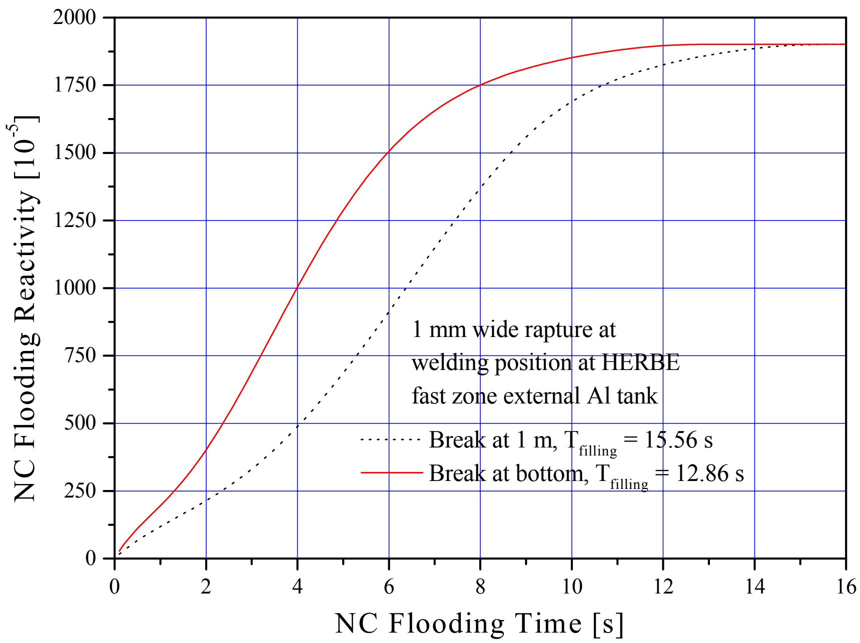

Figure 9 NCZ flooding reactivity-time function

5.3 Accident approximations

Kinetic parameters of the coupled fast-thermal system are measured [19] or determined by computer code. Very fast reactivity change follows in short time of the accident duration, so a space independent module ALFA_AC from code MACAN is used for power excursion calculation with the system integral parameters. Beside the MACAN code, the well-known point kinetics code AIREK II [21] is used to obtain power and energy time dependence during the accident. The code is modified to include determined external reactivity-time function and safety rods reactivity-time function for the case of the accident in the HERBE system.

In some cases of accident analysis, model without reactivity feedback is used due to low temperature coefficients of reactivity (TCR) $-1.5\cdot10^{-5}$ K^{-1} for fuel and $-4.0\cdot10^{-4}\cdot K^{-1}$ for moderator [22], and due to large value of the moderator heat capacity. All calculations are performed under assumption that both of DCM are failed to activate the safety system. The reactor initial power of 10 mW is chosen, corresponding to the lowest DC from neutron chambers and to the longest response time of the RB safety system.

5.4 Results of the flooding accidental analysis

All calculations are performed for the worse case i.e. the break of the NCZ Al tank at the bottom, and due to low total energy developed in the first 60 s, only that time interval for reactivity-time and power-time functions is shown in the Figures 10 and 11.

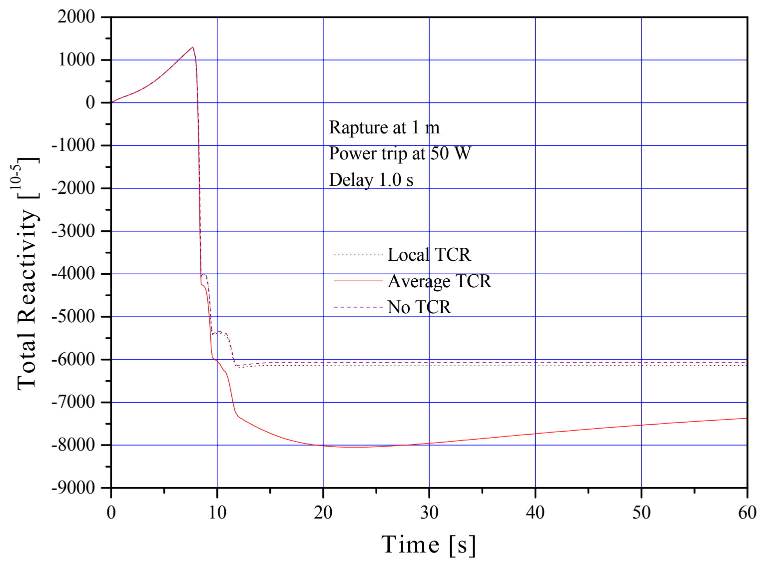

Figure 9. Reactivity-time function vs TCR model

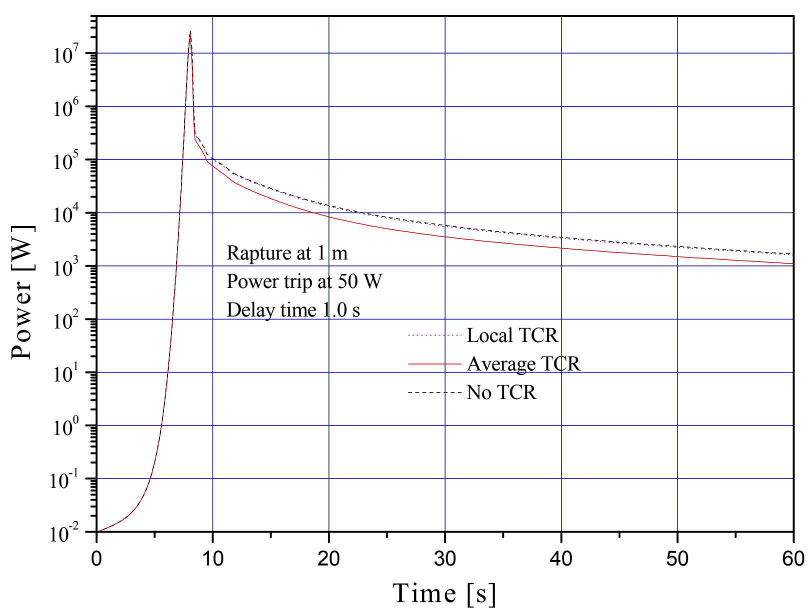

Figure 10. Power-time function vs TCR model

Three versions of the calculation are shown: (a) no TCR model, (b) core-averaged TCR model (MACAN and AIREK II code) and (c) proposed local (zones) TCR model. Calculations of power excursion in the case of the NCZ flooding are performed in wide range of assumed delay times and possible power thresholds for both of case of assumed rapture of the NCZ external Al tank [20]. It was shown (Figure 11) that even in the case of delay of 3.0 s after the safety system is triggered by the power threshold set at 20 mW (4.87 s after beginning of the accident) the HERBE system can be shut-down safely without any damage to the its components. Reactor power peak of 52 kW is reached after 5.50 s after the accident beginning. In the first minute of accident the total energy of 19 kJ is developed what is not enough to increase moderator temperature even for 1 K. Total equivalent dose in the reactor building, at the most exposed 'dosimetry point' (# 6 in the north corridor), in the first minute will be less than 8.5 mSv. All the other cases of the accident analyses shows slower power rate and/or lower power peaks.

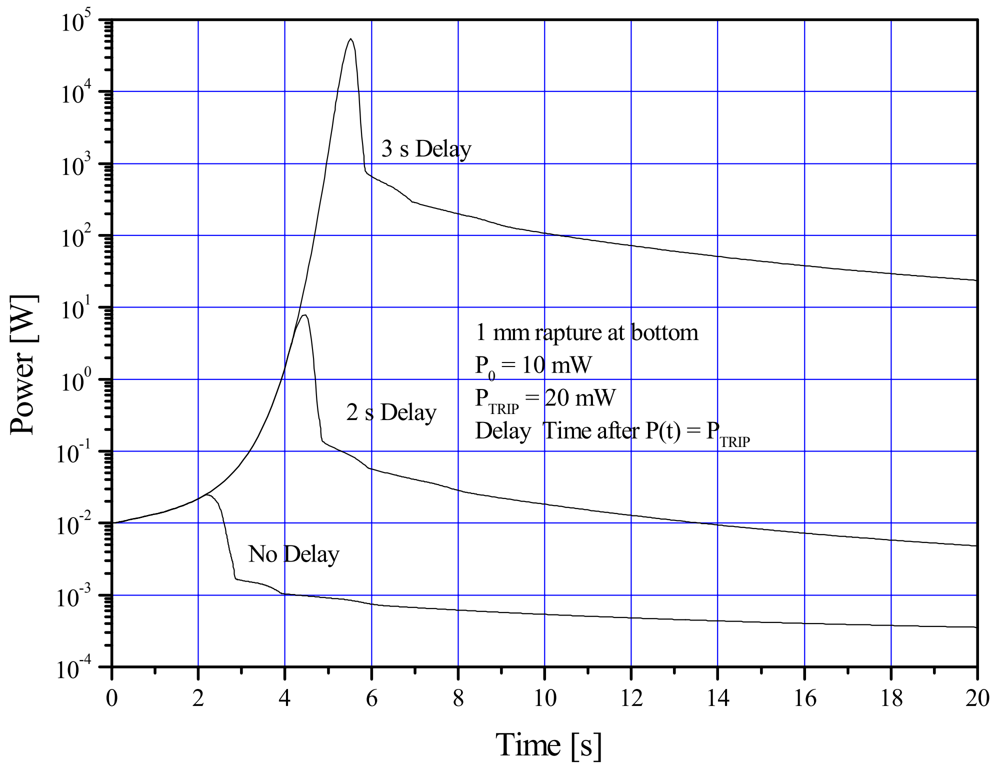

Figure 11. Power-time vs delay of HERBE safety system

6. BRIEF DESCRIPTION OF COMPUTER CODES

MACAN code [10] is reactor kinetics and dynamics code developed in the Vinča Institute of Nuclear Sciences, Yugoslavia with aim to be used in accidental analyses of the heavy water RA and RB research reactors that use tubular TVR-S type uranium fuel elements. It was verified, together with point kinetics codes SCM and AIREK II, at experimental data available for RB power excursion accident that was occurred in 1958. The code is also used for analyses of assumed accidents at both research reactors in the Vinča Institute during preparations of the PSARs and FSARs.

Difficulties in reactivity calculations in coupled fast-thermal core reactors are consequences of different neutron spectra in the cores and intensified local effects. A modified space-independent reactor kinetics model with space dependent feedback reactivities in accident analysis of the HERBE is developed [20] and included in the MACAN code. This quasi-space-dependent kinetics model, similar to classic nodal method, includes same equations for neutron population, delayed neutron and photoneutron precursors concentrations as in the space-independent model. But, equations for change of temperatures of fuel and coolant are space-dependent (in zones) as well as temperature coefficients of reactivity (TCR). The same model is adopted for feedback reactivity due to steam (void) generation in the coolant. These equations are set according to various fuel type (U metal or 80% enriched UO_2 dispersed in Al), geometry (full rod or annular cross-section) and coolant or moderator (D_2O or air) used in the 'HERBE' design. Total reactivity during reactor dynamics is a sum of external reactivity, reactivity of safety system and feedback reactivity. The feedback reactivity is defined as sum over space (in radial zones, superscript 'z') depending of fuel type (index 'f') and coolant type (Fig. 4, part C: index 'c1', 'c2').

$$\rho_{FB}(t) = \sum_{z=z_{fueltype1}} \left\langle \alpha_f^z \delta T_f^z(t) + \alpha_m \delta T_m(t) \right\rangle + \sum_{z=z_{fueltype2}} \left\langle \alpha_f^z \delta T_f^z(t) + \alpha_{c1}^z \delta T_{c1}^z(t) + \alpha_{c2}^z \delta T_{c2}^z(t) \right\rangle$$

The coupled fast-thermal core system is divided in 8 radial concentric material zones (nodes) in which the proposed void and temperature coefficients of reactivity are calculated using lattice cell and reactor computer codes. These reactivity models with appropriate heat transfer correlations for natural convection in narrow channels are implemented in MACAN code. System of differential equations is solved using either Runge-Kutta method of 5^{th} order or Hamming predictor-corrector method. Mass and energy balance is verified at every step of calculation in the code.

SCM code [11] is space-independent (point) kinetics code based on stiffness confinement method. The code is developed in the Vinča Nuclear Engineering Laboratory for fast accidental analyses. Applying the Stiffness Confinement Method (SCM) in numerical integration of the point kinetic equations it was possible to apply bigger time steps than in the usual standard methods, without including additional approximations or missing generality. Generally, time step for numerical integration of kinetic equations should be less than $10*\Lambda$ (Λ is prompt neutron generation time). It was shown that SCM allows achievement of the same accuracy as the standard integration methods can achieve, but the time steps in the SCM could be up to three orders of magnitude higher than the prompt neutron generation time. In such a way computing time required for the solving of kinetic equations was significantly decreased, what had enabled application of the code in real time reactivity measurement and prompt display of evaluated data at the operator control panel, i.e., at screen of the computer monitor.

7. CONCLUSION

In this paper, a brief overview of some accidents, methodology and computation tools used for the accident analyses at RB reactor are shown. A special attention is given to experiences collected in analysis of accidents in the coupled fast-thermal cores.

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