REVIEW OF ACCIDENT ANALYSES OF RB EXPERIMENTAL REACTOR

by

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Received on August 20, 2003; accepted in revised form on September 8, 2003

The RB reactor is a uranium fuel heavy water moderated critical assembly that has been put and kept in operation by the VINČA Institute of Nuclear Sciences, Belgrade, Serbia and Montenegro, since April 1958. The first complete Safety Analysis Report of the RB reactor was prepared in 1961/62; yet, the first accident analysis had been made in late 1958 with the aim to examine a power transition and the total equivalent doses received by the staff during the reactivity accident that occurred on October 15, 1958. Since 1960, the RB reactor has been modified a few times. Beside the initial natural uranium metal fuel rods, new types of fuel (TVR-S types of Russian origin) consisting of 2% enriched uranium metal and 80% enriched UO_2, dispersed in aluminum matrix, have been available since 1962 and 1976, respectively. Modifications of the control and safety systems of the reactor were made occasionally. Special reactor cores were designed and constructed using all three types of fuel elements, as well as the coupled fast-thermal ones. The Nuclear Safety Committee of the VINČA Institute, an independent regulatory body, approved for usage all these modifications of the RB reactor on the basis of the Preliminary Safety Analysis Reports, which, beside proposed technical modifications and new regulation rules, included safety analyses of various possible accidents. A special attention was given (and a new safety methodology was proposed) to thorough analyses of the design-based accidents related to the coupled fast-thermal cores that included central zones of the reactor filled by the fuel elements without any moderator. In this paper, an overview of some accidents, methodologies and computation tools used for the accident analyses of the RB reactor is given.

Key words: accidents, final safety analysis report, RB reactor, HERBE

INTRODUCTION

The RB reactor [1] is an unshielded critical assembly designed in 1958 to operate using natural (metal) uranium fuel rods in heavy water. In 1962, 2% enriched uranium metal fuel of TVR-S type (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 additionally. Such 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 Hybrid RB Experiment – a coupled fast-thermal core HERBE (1990), were designed in latter years. Simultaneously, the Operation and Regulation Rules and the Safety Analysis Reports were updated.

In most of about 4100 experiments carried out up to nowadays, the RB critical assembly has been operating as a pure thermal heavy water reactor at thermal power levels in range from 10 mW to 50 W. The initial reactor core was loaded with natural uranium (metal) fuel elements designed as rods covered by 1 mm thick aluminum cladding (April 1958).

A power excursion accident, in which six staff personnel were irradiated heavily (one died after few days), occurred at the reactor only six months after the first start-up. The initial accident analysis was done by a simple assumption of a power excursion

Review paper
UDC: 621.039.512/.519
BIBLID: 1451-3994, *18* (2003), 1, pp. 3–15

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according to an exponential law with a fixed reactor period. The accident analysis of the RB reactor was made in 1962 by using simple, space-independent codes, run on an analogue [5] or a digital computer [6]. More lately, in the beginning of nineties, this power excursion accident was analysed in more details [7] using more appropriate computer codes. The basic description of this accident and results of the power excursion analyses are given in the next section of the paper.

Since 1962, a safety assessment of the reactor operation with different cores has become a regular practice. The first Final Safety Analysis Report (FSAR) and the Regulation Rules were written in 1962, when the reactor control, safety and dosimetry systems were modernised to allow the reactor to operate with 2% enriched uranium metal fuel elements of TVR-S type. Further refurbishments of the reactor control equipment were made 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. The modifications, mentioned above, have converted the RB critical assembly to a flexible experimental reactor with 1 W nominal power. Since then, the RB reactor has been operating usually at power levels in range from 10 mW to 50 W, and, in special occasions, at very high power, up to 10 kW in a short time interval.

Among various thermal cores designed and examined in the RB reactor, the core No. 5/1973 designed in 1973 for the irradiation purpose by neutrons outside the reactor tank requires the special attention. Such a core of the RB reactor, with a 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*. The reactor control system was modified to allow operation of the reactor up to a power level of few kilowats. The experiment was performed without any problems, but a recent analysis [8] has shown that the reactor operated at approximately 2.5 times higher power level than declared by the operating staff. It was a consequence of a fact that reading instruments for the fission power were not calibrated for the new core configuration and the safety analyses and operation rules were not studied in full details.

Initial studies of design of fast neutron fields behind the reactor tank or inside it started when 80% enriched uranium fuel elements became available. Various special experimental cores have been designed up to now. The safety analysis report for the operation of the RB thermal cores using highly-enriched uranium (HEU) TVR-S fuel elements was updated in 1977/1978 when two new start-up channels with proportional BF_3 counters were included in the control system.

The INC is a thermal-to-fast neutron flux converter designed inside the RB tank. The fast zone (without moderator) of the INC-1 was designed as an annulus filled with 80% enriched fuel elements and surrounded with the blanket zone, made of two layers of natural uranium metal fuel rods in three separate aluminum tanks. A central air hole was designed for irradiation purposes. Three different versions of the INC were developed with various positions of the fuel elements in the fast zones: INC-1, INC-2 [2], and INC-3 [4]. The INC thermal zone was designed as the RB thermal core of 2% and 80% enriched fuel elements placed in a square lattice with pitch of 12 cm, surrounded with heavy water reflector. The recent INC-3 version was designed as a single axial ring using 80% enriched uranium fuel elements and thin cadmium layer in the fast zone, offering a large space (30 cm diameter, 120 cm deep) for irradiation purposes by fast neutron flux.

The HERBE system [3] is a coupled fast-thermal neutron core designed in the RB reactor tank with the aim to increase intensity of fast neutron flux in the vertical experimental channel placed in the centre of the fast core. Characteristics of the HERBE system were determined by computer codes and by the experiments carried out for verification purposes. Brief description of the HERBE system from a safety point of view is given in this paper. New FSAR and appropriate changes in the Operation and Regulation Rules for the RB reactor with HERBE system were written in 1991. The FSAR of the HERBE included thorough safety analysis of assumed accidents. Finally, basic descriptions of a few assumed accidents and results of power excursion study of the HERBE system are given, as well as an analysis of the most dangerous accident of the HERBE system – flooding of the fast zone by moderator – accompanied by results of new, recent measurements of the reactivity-time function of the RB reactor safety system.

ANALYSIS OF RB REACTOR 1958 ACCIDENT

Short description of the accident

The accident occurred on October 15, 1958, during an experiment carried out at the RB critical assembly, fig. 1. The staff personnel operating the reactor were in the reactor hall, fig. 2. The heavy water core (square lattice pitch 12 cm) was loaded with 208 natural uranium metal rods forming an un-reflected ("bare") critical system. Descriptions of the accident are given in a few references [9–15],

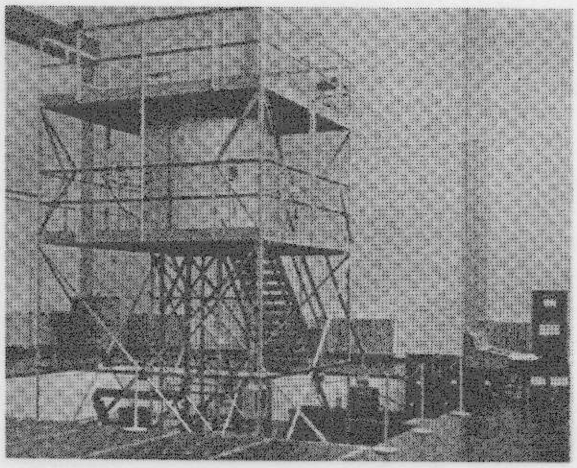

Figure 1. View of the RB reactor in 1958

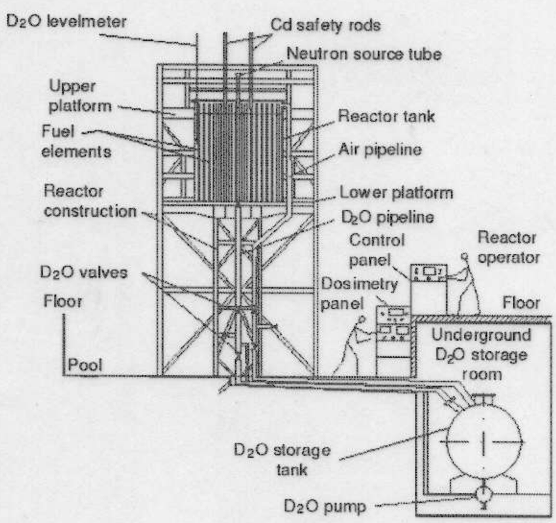

Figure 2. Sketch of RB staff positions during 1958 accident

including the very first reports on accident consequences at the irradiated staff [6, 16].

At heavy water level of 175 cm (3.5 cm below the expected critical level) in the core, a fast increasing of the moderator level (2.5 cm per minute) was switched on with the aim to achieve new expected sub-critical level of 177 cm. During that process the staff operating the reactor was disturbed by a non-staff person entering the reactor hall. The heavy water reached a level of 177 cm in the reactor core and continued to increase, because the operator did not switch off the pump. Safety designed instrumentation of the RB reactor, used in dosimetry, radiation level alert and safety systems, was either switched off or removed partially. After 84 s (counting from the time moment when the heavy water pump was switched on, $t = 0$), the critical level (178.5 cm) was reached and further increase of heavy water turned on the reactor to become super-critical. The reactivity and reactor power continued to increase without any supervision of the staff.

Since the pump was not switched off, the whole amount of heavy water was transferred from an underground storage tank into the reactor tank. The total excess of heavy water was 4.5 cm above the critical level. Two BF_3 counters, used by experimenters and operators in the reactor hall, believed to work properly, had reached saturation level and were reading a constant maximum value even though the power was increasing steadily. The third BF_3 counter, behaving erratically, was disconnected and help of the maintenance team was asked for. An automatic paper recorder used for measuring airborne activity and radioactive fallout, belonging to the equipment installed at a roof of a building, about 540 m away from the RB reactor building, registered the power rise and accompanying increased gamma-ray background level for a time interval of approximately 10 minutes.

Time duration of the power excursion was not recorded in the reactor logbook. Results of the new calculation show that the RB assembly was in the non-controlled state, at heavy water level of 183 cm, for 433 seconds, when the staff personnel smelled ozone due to ionised air in the reactor hall. Only then did the staff realise that the system was supercritical and one staff member shut down the assembly manually by using two cadmium safety rods. All six personnel operating the reactor were exposed to high levels of radiation originating from neutrons and gamma-rays. On the day following the accident, the irradiated personnel were transferred by airplane to the Marie Curie hospital in Paris. Equivalent doses, received by the personnel, were estimated at levels of about 50% of the lethal dose. One staff member died few days later, in spite of the best available medical treatment that, among other methods, included bone marrow transplantation. Many years latter, in an non-edited personal paper [17], written by a staff member participating in the accident, a few facts, already assumed earlier, were revelled and confirmed; among them was the fact that the operating staff was familiar neither with the basic reactor physics, reactor functional parts, and regulation rules, nor with the possibilities and consequences of potential accidents. Inspection of the fuel elements in 1961 showed that about 30% of fuel rods had signs of small bubbles or bending at the aluminum cladding.

Accident analyses

According to the absolute activity determined by measuring irradiated gold and copper foils found in the RB building and some metal objects, carried by the irradiated employees, it was estimated [9] that the total fission energy generated in the acci-

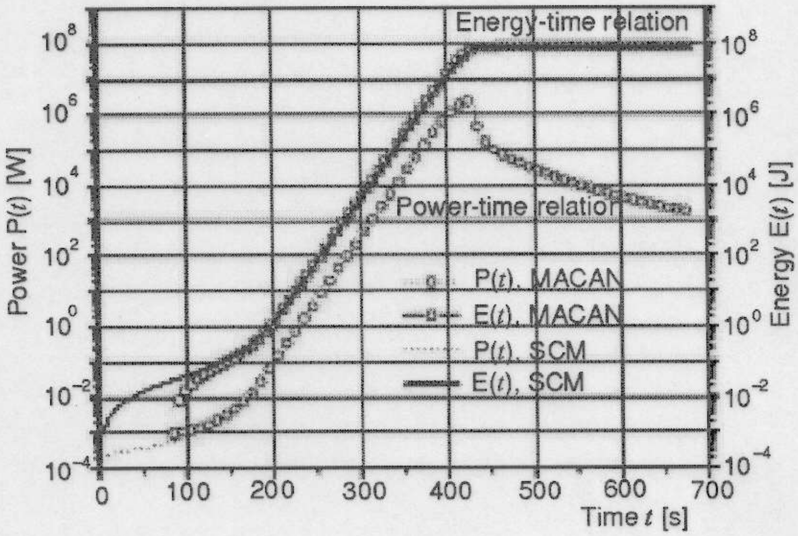

Figure 3. Power and energy *vs.* time in RB 1958 accident

dent was 80 MJ (~$2.6 \cdot 10^{18}$ fission). The first analyses of the RB reactor 1958 accident were done in 1958 [10, 11] and 1960 [6], based on a simple approximation of the power excursion by an exponential function with a 10 second period. Recent calculations [7] carried out by using MACAN [18] and SCM [19] codes have shown that the RB reactor, at heavy water level of 175 cm, was sub-critical with reactivity of $-0.305\beta_{\mathrm{eff}}$. For heavy water excess of 4.5 cm above the critical level, the system was supercritical with reactivity of $0.375\beta_{\mathrm{eff}}$. The power was steadily increasing within a period of 12.3 second and reached the maximum level of nearly 2.5 MW, with the total released energy of 80 MJ, fig. 3.

SAFETY ASSESSMENT OF IN-CORE FAST NEUTRON FIELDS OF RB REACTOR

All versions of fast neutron fields at the RB reactor were designed and constructed by research staff of ex-Nuclear Engineering Laboratory (now Centre for Nuclear Technologies and Research – Centre NTI) of the VINČA Institute of Nuclear Sciences. Safety aspects of this development were monitored, reviewed and licensed by the Nuclear Safety Committee, an independent expert body of the VINČA Institute. The PSARs, FSARs and the Operation Rules were also reviewed and approved for usage by the Nuclear Safety Committee, and proposed to the Director General to issue permission (the licence) for a regular operation of the reactor with each of the new fast neutron fields.

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, without significant modifications, can be used,

(c) the whole neutron coupling (RB reactor thermal core – fast neutron field) should be "strong" in such a way that the coupled system will operate as a common thermal reactor (with large prompt neutron lifetime) so that the existing control system of the RB reactor can operate normally, and

(d) the neutron coupling should be designed in such a way that the coupled system can be shut down quickly and safely using the safety rods dropping in thermal core only.

All these demands, set for design of the fast neutron fields at the RB reactor, were achieved in the construction phase and verified in operation. The safety analyses have shown that operation of the RB reactor with designed fast neutron fields is safe, without the need for any significant modification of the control or safety systems. The existing system of the safety rods has enough (negative) reactivity, which can be inserted in a very short time interval, to stop any assumed power excursion timely and safely. Response times of the reactor control equipment are such that all control of the designed neutron couplings can be done 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 either the most probable accident or in a case of an accident in which the highest reactivity is inserted. Neither the reactor system components nor the reactor staff should be exposed to the high doses of neutrons and gamma-rays (higher than 25 mGy) during these assumed serious accidents. In order to increase the sensitivity of the safety system to the most dangerous possible accident (*i. e.*, the flooding of the fast zone of the INCs and HERBE by heavy water from the thermal core), two moderator leak sensors (DCM) are placed in the outermost aluminum tank of the fast zone and independently connected to the RB reactor's existing safety chain.

Recently, the RB reactor has operated with the special cores designed as the third version of the INC, INC-3 [4] and with new-designed large-hole irradiation facility [20]. These special cores are used mainly for irradiation purposes. Also, these facilities are used for studies of development of modern systems used for radiation protection, reactor control and safety, and for verification of new computer codes for reactor design and safety analyses, developed at the Centre for Nuclear Technologies and Research in the VINČA Institute. Another fields of the RB reactor's recent applications are nowadays increasing interest for (1) the compilation and systematisation of the evaluated benchmark experiments in the criticality safety and (2) sub-criticality measurement studies useful in a design of Accelerator Driven Sub-critical Systems (ADS). Three separate evaluations of more than 20 carefully selected, well-documented and reviewed RB reactor criticality experiments are included in the International Criticality Safety Benchmark Evaluation Project (ICSBEP) handbook, managed by the Nuclear Science Committee (NSC) of the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) [21]. The project is led by the Idaho National Environmental and Energy Laboratory (INEEL), operating for the Department of Energy (DOE), USA, while the handbook is issued each year, as the CD ROM edition and the Web presentation at the INEEL web site.

Description of the HERBE coupled fast-thermal system

The "Fast Zone" (FZ) is a region of the HERBE system formed as a three-zone dry fuel core in the centre of the RB reactor. The central zone of the FZ is the "Fast Core" (FC) designed of natural uranium metal fuel elements in the first (the innermost) aluminum tank (200/202 mm diameter) with axial vertical experimental channel (VCH). The fast core is surrounded by the "Neutron Filter Zone" (NF). The NF is designed using 1.6 mm thick cadmium foil and natural uranium metal fuel elements in the second (middle) aluminum tank (300/302 mm diameter). The "Neutron Converter Zone" (NC), which surrounds the FC and NF, is formed using 80% enriched UO_2 fuel elements in the third (the outermost) aluminum tank (400/408 mm diameter). Each aluminum tank is closed at the bottom and waterproof, so there is no moderator in the fast zone. The total height of the fast zone is 140 cm. Thermal core (driver) is designed using 44 fuel elements with 80% enriched UO_2, placed in 12 cm square lattice pitch and filled with heavy water moderator and reflector. Neutron coupling zone between the fast and the thermal regions is formed of 7 cm thick heavy water. Radial reflector of the heavy water surrounds the thermal core in the RB reactor tank (100/101 cm diameter). Cross-sections of the HERBE coupled fast-thermal system and the enriched uranium TVR-S fuel element are shown in figs. 4 and 5 respectively.

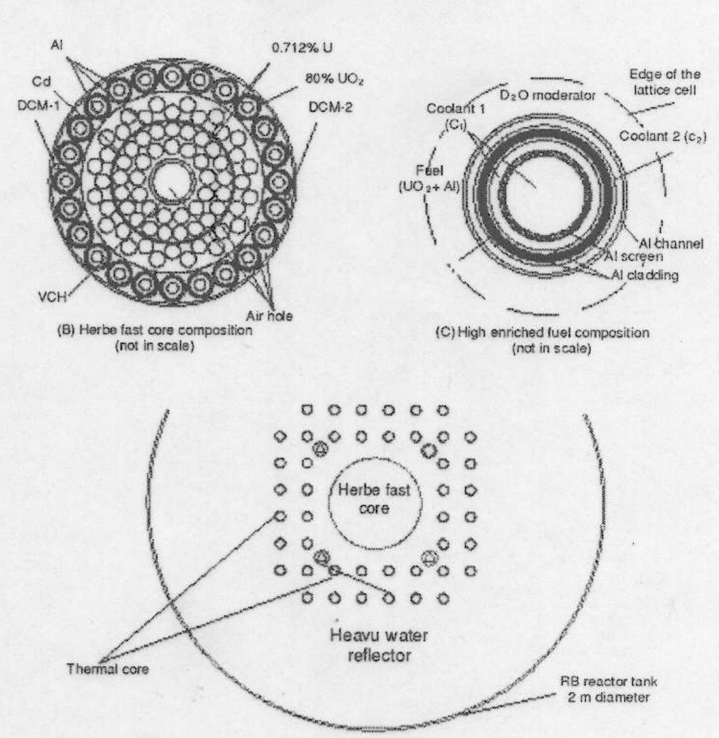

Figure 4. Horizontal cross-section of the RB coupled fast-thermal system HERBE

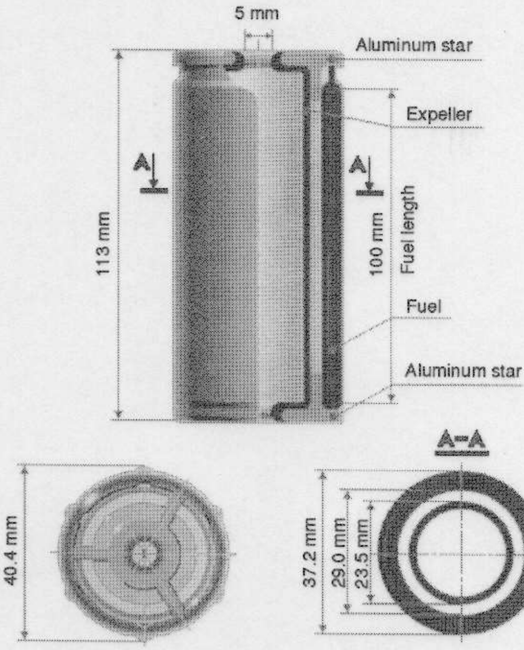

Figure 5. TVR-S slug cross-section

RB reactor safety system basic characteristics

The RB reactor power has been monitored, since 1987, by nine gamma-ray compensated neutron sensitive ionisation chambers of very low power level (10 mW) corresponding to the intensity of direct current (DC) of 1-10 pA, depending on particular neutron chamber sensitivity. The response time of the instrumentation of the reactor power channels at such low DC (*i. e.*, at the reactor criticality level) is relatively long (1.5 s for the linear DC power channels, 1.0 s for the logarithmic DC power channel, and 4.0 s for associated period-meters). The RB reactor safety system is based on the safety chain designed at the safety logic "one of two". In such a way the RB reactor operation is stopped (trip) by triggering any of 18 safety thresholds (ST) connected to the closed chain of the safety system:

(a) AC electric power failure (3 ST) and DC electric power failure trip (1 ST),

(b) linear power channel overpower trip (3 ST),

(c) logarithmic power channel overpower (3 ST) or minimum period (3 ST) trips,

(d) dosimeter channel gamma-ray overdose rate trip (1 ST),

(e) linear and logarithmic power recorder overpower trip (2 ST), and

(f) moderator leaking sensors (DCM) in the HERBE fast zone trip (2 ST).

As the result of the safety system activation, all four rods drop into the reactor core and an increase of heavy water level in the reactor tank is stopped, if switched on.

ANALYSES OF SAFETY RELATED EVENTS IN HERBE SYSTEM

The majority of the safety analyses of the HERBE system is based on the data and conclusions given in the RB reactor's previous safety analysis reports and on more than 30 years of operational and safety experience of the reactor and safety personnel. All safety analyses are presented in full details at the HERBE FSAR [22]. During operation of the HERBE coupled fast-thermal system, the following events are accepted and analysed as main causes of possible incidents and accidents: (a) increasing of heavy water level over the critical level; (b) control rod (*i.e.*, the safety rod SR3) withdrawing at the criticality; (c) sudden filling of an experimental channel in the thermal core with heavy water moderator; (d) moving out an experimental or other part of equipment from the thermal core, and (e) sudden flooding of the fast zone with heavy water moderator from the thermal core.

Increasing of heavy water level over the critical level

The criticality of the RB reactor is attained at steady level and all power changes are performed by the operator's manual actions, changing the heavy water level in small increments or decrements. Thus, increasing of the heavy water level over the critical level is recognised as the most likely event leading to an accident due to the possibility of the operator's error during operation of the RB reactor. An interlock system is designed in such a way that moderator increasing is turned off automatically after every 60 s, resulting in maximum increase of the heavy water level of only 8 mm. The heavy water gradient at the critical level was measured as $(191.6 \pm 1.5)\cdot 10^{-5}$ cm^{-1}, which was very near to the calculated one, determined as $195\cdot 10^{-5}$ cm^{-1}. The increase of heavy water can be represented as a reactivity-time ramp function with the reactivity rate of $2.6\cdot 10^{-5}$ s^{-1}. It is a very slow reactivity rate easily controllable either by the operator's action or by an automatic action of the safety system of the RB reactor. Only in the case of a partial failure of the RB safety system and great carelessness of the reactor staff in the control room can this slow increase of heavy water drive the reactor power to the high level.

Withdrawing of safety rod at critical level

The RB reactor interlock system is designed in such a way that simultaneous withdrawing of two safety rods is not possible. In case of the HERBE system, the low-reactivity control rod was replaced by new-designed high-reactivity safety rod SR3. During the operation procedure of approaching the criticality, the SR3 is last withdrawn from the reactor core at sub-criticality of the $k_{eff} \sim 0.8$. This standard operation has a possibility to result in an accidental situation only if the actual value of the k_{eff} is mistakenly judged to be lower than it really is, *i.e.*, very close to 1.0. This event can be recognised easily either automatically by the RB reactor power channels included in the safety system or by manual actions taken by the staff in the control room.

Filling of an experimental channel with moderator

The RB reactor is designed in such a way that the different experimental channels can be placed in the core. Sudden rapture in a vertical experimental channel (with maximum possible diameter of 100 mm, placed at the highest reactivity position in the thermal core) can be represented as a reactivity--time ramp function with the reactivity rate of $31 \cdot 10^{-5}$ s^{-1} in the first 10 s, during which the heavy water moderator floods the experimental channel completely. It has been shown that the reactor safety system can safely stop the reactor power excursion in that case.

Moving out the experimental equipment from the reactor core

According to the RB reactor's Operation Rules, the experimental equipment (detectors, samples, *etc.*) can be placed in the reactor core only if its reactivity is less than $200 \cdot 10^{-5}$. Moving out the equipment during the RB reactor operation at the critical level (at approximately 10 mW) can be performed only under a supervision of the reactor staff. It has been shown that the safety system can safely stop the reactor in the case of such an event leading to an accident.

FLOODING OF HERBE FAST ZONE WITH HEAVY WATER

It has been accepted that there is a significant probability that the external aluminum 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. For example, the event could be the consequence of a sudden strong earthquake. The situation enables a penetration of the moderator from the thermal core into the HERBE fast zone and results in the high and fast reactivity increase of the entire coupled fast-thermal system. For that reason, the maximum of the safety related philosophy is applied in the design and construction of the HERBE:

(a) three separate aluminum tanks for the fast zone, each closed at the bottom, are designed and checked for the water leaking and the welding quality,

(b) HEU fuel elements in the NC are placed in the aluminum "fuel channels" sealed at the bottom so that the moderator could not penetrate within,

(c) HEU fuel elements in each "fuel channel" in the NC are replaced at the channel bottom by hollow aluminum supporters (43.0 cm long) closed in such a way that the heavy water could not penetrate within,

(d) two separate moderator-leaking detectors (DCM) are placed into the NC and connected independently, *i. e.*, at different places, to the reactor safety chain, and

(e) low-reactivity control rod is replaced by new-designed high-value reactivity rod SR3.

It has also been assumed that breaking more than one aluminum tank at the same time (of the existing three ones forming the HERBE fast zone) has no significant probability for safety analysis. On the other hand, the cadmium layer in the NC zone would act as the strong neutron absorber, not allowing a significant increase of the external reactivity during the flooding of the FC zone. Determinations of the true reactivity-time dependence during flooding of the NC and timely action of the HERBE safety system in the accident analyses were of major importance during the design stage. For the verification purposes of the accident analyses results, the specific safety experiments [22–27] were carried out at RB reactor with HERBE core in order to determine the reactivity of the reactor safety rods and their drop-in times. Additionally, the special experiment [22, 23, 29] with the controlled flooding of the NC zone by the moderator was done with the aim to verify the results of the calculations.

Determination of the HERBE safety rods reactivity-time function

HERBE system is designed with 4 safety rods. Two rods are the "regular" safety rods (SR1 = SS1 and SR2 = SS2), which, after the activation of the safety system, drop into the core in a short time, without any pause during motion. The third safety rod (SR3 = SS3) is designed instead of the previously used control rod and, after triggering of the safety system, it drops into the core with one cessation during the motion. The heavy water level meter (WLM = PN) acts in the safety system as the fourth safety rod (SR4) that drops into the core with two cessations during the motion.

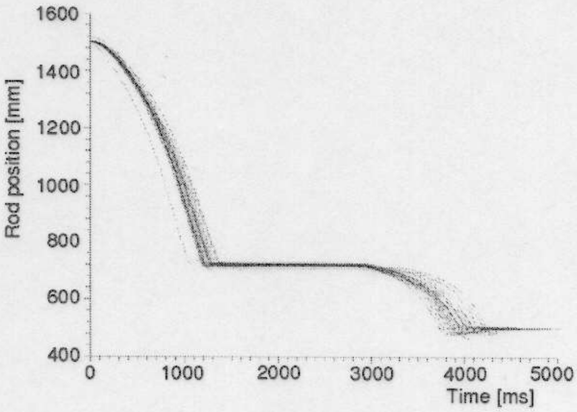

Figure 6. Measured trajectories of the SR3

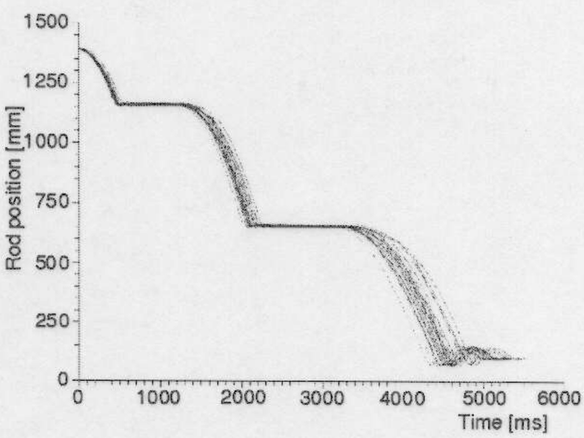

Figure 7. Measured trajectories of the SR4

The accurate determination of the rod drop times was requested by the Nuclear Safety Committee and was carried out in the specific experiments in the HERBE system. In the initial experiments

Figure 8. Measured motion laws for safety rods of HERBE system

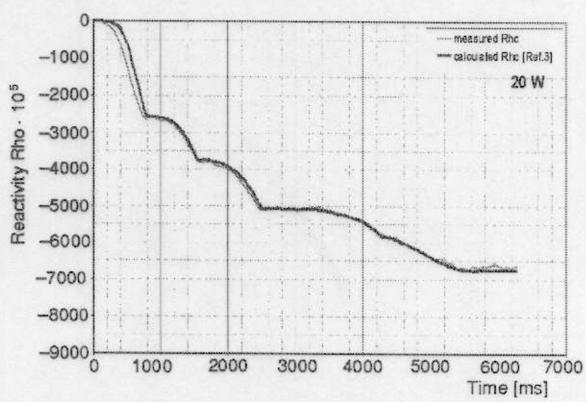

Figure 9. Reactivity-time functions for HERBE safety rods

[24], an electromagnetic transducer of the rod's motion into voltage and the voltage-to-time conversion by an analogue-to-digital converter (ADC) in a computer were applied. Evaluation of the experimental data, under assumptions of the constant acceleration during the motion and instantaneous interruption of the motion, has shown a very complex timing of the HERBE safety rods. In the later phase [25], the measuring equipment was improved and a new digital optical incremental device was used for on-line data measurement of the functions of the position-time for the safety rods SR3 and SR4 that had cessations during the dropping. Results of the measured trajectories are shown in figs. 6 and 7, respectively. Averaged measured trajectories (the "motion law") for all four rods of the RB reactor safety system are shown in fig. 8. One could see, from the curves presented in fig. 8, that the safety rod SR1 did not pass the whole travel path during dropping. It stacked at approximately half way to the bottom position. This event is not good from the safety point of view, but the measurement results show that this measuring technique could be used for on-line monitoring of the operation of the safety or control rods. Any unexpected event during rods' motion could be detected immediately and the data may be forwarded to the maintenance team for a repair.

The reactivity worth of the safety rods of the HERBE system was determined in a series of measurements and compared to the calculated ones. In the experiments, each safety rod was inserted in the core either separately or in a combination with the others. Neutron time distribution during shutdown was recorded, both in the multi-channel analyser (MCA) using a proportional BF_3 counter, and in a PC by digitalising (by an ADC) the current output from a DC amplifier connected to a neutron sensitive chamber [26]. After smoothing the data, the reactivity was determined from the known neutron flux-time distribution using the IM code [27] based

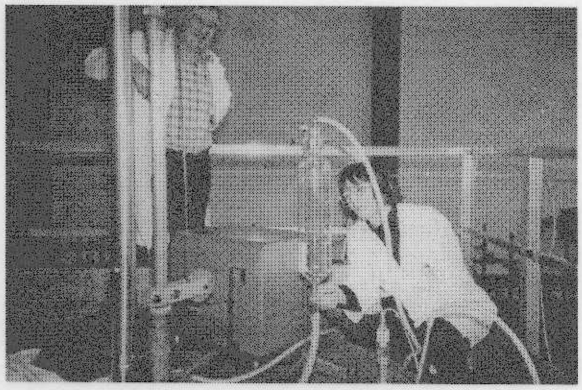

Figure 10. Operating of the decanting device at the reactor top

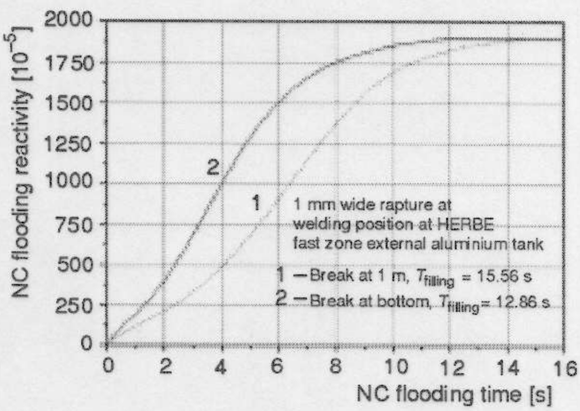

Figure 11. Calculated reactivity-time function for NC flooding

on the Inverse kinetics method. The kinetics parameters – the effective fraction of delayed neutrons and photo-neutrons (β_{eff}) and the prompt neutron generation time (Λ) – of the HERBE system were determined using computer codes AVERY [27] and VEGA [28] and verified in the separate experiments [30]. The normalised reactivity of the water level meter (WLM), as a function of the normalised rod position in the HERBE system, was approximated by the polynomial function of 5^{th} order determined by the best linear fit of the experimental data. This result was used in the final computational determination of the complex function of reactivity (Rho) *versus* time of the safety rods. The calculated function is compared to the measured one [26] and shown in fig. 9.

External reactivity-time function during NC flooding

All possible activities in the HERBE construction phase were done to prevent the heavy water from thermal zone entering the fast zone in the case of an accidental flooding. These construction details have reduced the total reactivity excess for almost 50% and have also reduced the reactivity insertion rate to the acceptable value controllable by the safety system of the RB reactor. With the aim to verify the calculation result for this reactivity value, the special experiment with the controlled flooding of the NC was carried out. The decanting device was designed for the semiautomatic transfer of the moderator from the HERBE thermal core into the NC and installed and operated manually at the reactor top cover (fig. 10).

Due to the safety precautions, the heavy water transfer operation was carried out at the reactor being highly sub-critical. For each heavy water level in the NC, the critical level of the system was measured by the standard criticality approach procedure. Results of the experiments and initial calculations [3] have shown acceptable agreement and have also been confirmed by the recent calculations [31] carried out by the MCNP code [32]. A simple flooding model was applied [33], giving estimation of 13–16 s for the filling time of the NC with the moderator. To obtain this result, an instantaneous break of the 4 mm thick external aluminum tank for 1 mm width around the whole circumference of the tank was assumed. The increase of the heavy water level in the NC during the flooding time was used to calculate the change of the reactivity with the increasing of the height of the flooding moderator in the NC, determined by using computer codes. The "external" reactivity-time function, during the heavy water flooding time of the NC, calculated by codes and experimentally verified, is shown in fig. 11.

Accident approximations

Kinetics parameters of the coupled fast-thermal system are measured [30] or determined by computer codes [3]. Change of the system reactivity is very fast during short time of the accident duration, so the space-independent module ALFA_AC from the code MACAN is used for the power excursion calculations using the system integral kinetics and thermo-hydraulics parameters. Beside the MACAN code, the well-known point kinetics code AIREK II [34] is used to obtain the power and energy time dependence during the accident. The AIREK II code is modified to include actual dependences of (1) the external reactivity-time function and (2) the reactivity-time function of the safety rods for the case of the studied accident in the HERBE system.

In some cases of the accident analysis, a model without the reactivity feedback is used due to the low temperature coefficients of the reactivity (TCR) $-1.5\cdot10^{-5}$ K^{-1} for the fuel material and $-4.0\cdot10^{-4}$ K^{-1} for moderator [33, 35], and due to large value of

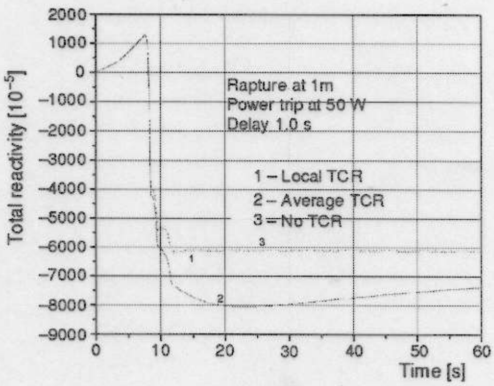

Figure 12. Reactivity-time function *vs*. TCR model

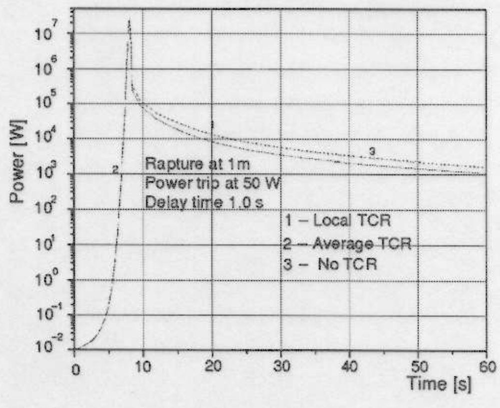

Figure 13. Power-time function *vs*. TCR model

the moderator heat capacity. All calculations are performed under the assumption that both of the DCM fail to activate the reactor safety system. The reactor initial power of 10 mW is chosen, corresponding to the lowest DC from the neutron chambers and, consequently, to the longest response time of instrumentation of the RB safety system.

Results of the flooding accident analysis

All calculations [33] have been performed for the worst case, *i. e.*, the break of the aluminum tank of the NC at the bottom. Due to the low value of the total energy developed in the first 60 s, only that time interval for the reactivity-time and the power-time functions is shown in figs. 12 and 13, respectively.

Three versions of the TCR model in the calculations are shown: (a) no TCR model, *i. e.*, TCR = 0; (b) the core-averaged TCR model (MACAN and AIREK II code), and (c) the proposed "local (zonal)" TCR model [33]. Calculations of the power excursion in the case of the NC flooding were carried out in the wide range of assumed delay times of the safety system and the power thresholds for both of the cases of the assumed rapture of the NC external aluminum tank [33]. It was shown (fig. 14)

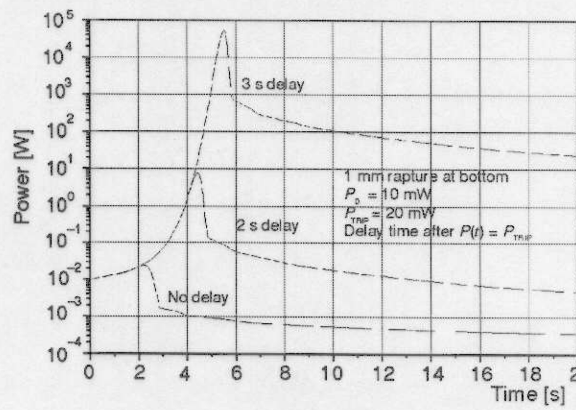

Figure 14. Power-time vs. delay of HERBE safety system

that, even in the case of delay of 3.0 s after the safety system was triggered by the power threshold set at 20 mW (4.87 s after beginning the accident), the HERBE system could be shut-down safely, without any damage to its components. The reactor power peak of 52 kW was reached 5.50 s after the start of the accident. In the first minute of the accident, the total fission energy of 19 kJ was developed, which was not enough to increase the moderator average temperature even for 1 K. The total equivalent neutron and gamma-rays exposition dose in the reactor building at the most exposed "dosimetry point" (No. 6 in the north corridor) in the first minute of the accident would be less than 8.5 mSv. All the other cases of the accident analyses show a slower power rate and/or a lower power peaks.

DESCRIPTION OF COMPUTER CODES

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

Difficulties in the reactivity calculations in coupled fast-thermal core reactors are consequences of different neutron spectra in the neutron coupled cores and intensified local effects. A modified space-independent reactor kinetics model with the space-dependent feedback reactivity in the accident analysis of the HERBE has been developed [33] and included in the MACAN code. This quasi-space-dependent kinetics model, similar to the classic nodal method, includes the same equations for the neutron population, delayed neutron, and photo-neutron precursors concentrations, as they are in the space-independent model. However, the equations for change of the temperatures of the fuel and the coolant are the space-dependent (*i. e.*, in zones) as well as the temperature coefficients of reactivity (TCR). The same model has been adopted for the feedback reactivity due to the steam generation in the coolant or due to existing of the void (*i. e.*, air) zones surrounding fuel elements in the fast zone. These equations are set according to the various fuel types (uranium metal or 80% enriched uranium-dioxide dispersed in aluminum), geometry (full rod or annular cross-section) and coolant or moderator (heavy water or air) used in the HERBE design. The total reactivity during reactor dynamics is the sum of the external reactivity, the reactivity of the safety system and the feedback reactivity.

The total feedback reactivity is defined as the sum of the "local feedback reactivity" over the space (in radial zones, superscript z) depending of the fuel type (index f) and the coolant type (fig. 4, part C: index c_1 and c_2).

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

$$+ \sum_{z=z_{fueltype2}} \left\langle \alpha_f^z \delta T_f^z(t) + \alpha_{c_1}^z \delta T_{c_1}^z(t) + \alpha_{c_2}^z \delta T_{c_2}^z(t) \right\rangle$$

The HERBE coupled fast-thermal core system is divided into eight radial concentric material zones (similar to the "nodes") in which, as defined above, void and temperature coefficients of reactivity are calculated using the lattice cell and the reactor computer codes. These reactivity models with the appropriate heat transfer correlations for conduction and the natural or forced convection in the narrow channels have been implemented in the MACAN code. The system of differential equations, formed in such a way for the RB and RA reactor' cores, is solved numerically using either the Runge-Kutta method of 5th order or the Hamming predictor-corrector method. The mass and energy balance is verified in the code at every step of calculation.

The SCM code is the space-independent ("point") kinetics code based on the Stiffness Confinement Method. The code has been developed in the VINČA Institute for fast analyses of accidental cases. Applying the SCM in the numerical integration of the point kinetics equations, it has been possible to use longer time steps than in the usual standard kinetics numerical methods, without including any additional approximations or missing generality. Generally, a time step for the numerical integration of the kinetics equations should be less than 10 Λ (Λ is prompt neutron generation time). It has been shown that the SCM allows achievement of the same accuracy as the standard integration methods, but the time steps in the SCM can be up to three orders of magnitude higher than the prompt neutron generation time. In such a way, computing time required for solving the kinetics equations significantly decreases, which enables application of the code in real time reactivity measurement and prompt display of evaluated data at the operator control panel, *i. e.*, at the screen of the computer monitor. The feedback reactivity is included in the SCM code by applying the "energy coefficient".

The AIREK II computer code is well-known space-independent ("point") kinetics code developed in the fifties [36] and modified latter in the sixties. It was widely used to evaluate kinetic behaviour of various power and reactivity transients.

CONCLUSION

An overview of selected accidents, methods applied and computation tools used for the accident analyses at RB reactor in the VINČA Institute of Nuclear Sciences, Belgrade, are shown in this paper. Various reactor cores were designed and constructed in the RB tank using three types of fuel elements, among them the coupled fast-thermal ones. The Preliminary and Final Safety Analysis Reports were prepared which, beside description of technical modifications and new regulation rules, included safety analyses of various possible accidents. A special attention in these reports was given to thorough analyses of the design-based accidents related to the coupled fast-thermal cores that included central zones of the reactor filled with the fuel elements without any moderator. In these accident analyses, during assumed flooding of the fast zone by heavy water moderator, a very high reactivity could be inserted in the system for very short time. In some cases, a new safety methodology had to be proposed and it was necessary to provide such a modified design of the safety system of the reactor that would have fast response to the accident and that would have enough high (negative) reactivity to shut down the reactor timely and safely.

ACKNOWLEDGEMENTS

The Ministry of Science, Technologies, and Development of the Republic of Serbia supported work on this topic through the Project No.

1958/2002, *Transport Processes of Particles in Fission and Fusion Systems.*

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Милан ПЕШИЋ

ПРЕГЛЕД АНАЛИЗА АКЦИДЕНАТА ЕКСПЕРИМЕНТАЛНОГ РЕАКТОРА РБ

Реактор РБ је уранијум-тешководни критични систем изграђен у Институту за нуклеарне науке „Винча", а у раду још од краја априла 1958. године. Први комплетан извештај о сигурносној анализи реактора РБ припремљен је тек 1961–1962. године. Међутим, прве анализе акцидента урађене су крајем 1958. године са циљем да се испита временско понашање снаге реактора и укупне еквивалентне дозе зрачења коју је примило особље реактора за време акцидента који се десио 15. октобра 1958. године. Након 1960. године, реактор РБ модификован је више пута. Поред почетног горива у облику шипки од природног уранијум-метала, набављено је ново гориво руског порекла у облику малих сегмената, тзв. ТВР-С елемената, од 2% обогаћеног уранијум-метала (1962. године), односно 80% обогаћеног уранијума (1976. године) у виду уранијум-диоксида дисперзованог у матрици од алуминијума. Модификације контролног и сигурносног система реактора РБ рађене су повремено. Од 1986. године, пројектована су и изграђена разноврсна специјална језгра реактора коришћењем сва три типа горивних елемената, као што су спрегнута брзо-термичка језгра. Дозволе за рад реактора РБ са овим сложеним језгрима издавао је директор Института на препоруку Комитета за нуклеарну сигурност Института „Винча", као независног експертског тела које је стручно анализирало припремљене Прелиминарне извештаје о сигурносним анализама, као и извештаје о предложеним модификацијама и све измене погонских Инструкција и Прописа о раду реактора. У овим извештајима посебна пажња посвећена је детаљним сигурносним анализама могућих акцидената, посебно оних везаних за пројектне акциденте неутронско спрегнутих језгара која укључују велике шупљине у центру језгра реактора. У овим анализама предложене су модификације постојећих модела кинетике и динамике реактора које су реализоване кроз рачунарски програм. Улазни подаци одређивани су најпре прорачунима, а потом верификовани у посебним експериментима. Анализом узрока могућих акцидената за спрегнуто брзо-термичко језгро ХЕРБЕ показано је да се у случају акцидента потапања брзе зоне модератором из термичке зоне у реакторски систем уноси врло велика реактивност за врло кратко време. Зато је било неопходно да се обезбеде такве пројектне модификације реакторског система које осигуравају смањење величине и брзине уношења спољашње реактивности, како би сигурносни систем имао довољно велику негативну реактивност сигурносних шипки и довољно време за брз одзив на тај акцидент, те да би реактор био правовремено и сигурно угашен. У овом раду дат је преглед сигурносних анализа и оцена неких изабраних акцидената реактора РБ као и елементи примењених модела, методологије и рачунарских програма који су у тим анализама коришћени.