APPLICATION OF BEST ESTIMATE PLUS UNCERTAINTY IN REVIEW OF RESEARCH REACTOR SAFETY ANALYSIS

by

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To construct and operate a nuclear research reactor, the licensee is required to obtain the authorization from the regulatory body. One of the tasks of the regulatory authority is to verify that the safety analysis fulfils safety requirements. Historically, the compliance with safety requirements was assessed using a deterministic approach and conservative assumptions. This provides sufficient safety margins with respect to the licensing limits on boundary and operational conditions.

Conservative assumptions were introduced into safety analysis to account for the uncertainty associated with lack of knowledge. With the introduction of best estimate computational tools, safety analyses are usually carried out using the best estimate approach. Results of such analyses can be accepted by the regulatory authority only if appropriate uncertainty evaluation is carried out. Best estimate computer codes are capable of providing more realistic information on the status of the plant, allowing the prediction of real safety margins. The best estimate plus uncertainty approach has proven to be reliable and viable of supplying realistic results if all conditions are carefully followed. This paper, therefore, presents this concept and its possible application to research reactor safety analysis.

The aim of the paper is to investigate the unprotected loss-of-flow transients "core blockage" of a miniature neutron source research reactor by applying best estimate plus uncertainty methodology. The results of our calculations show that the temperatures in the core are within the safety limits and do not pose any significant threat to the reactor, as far as the melting of the cladding is concerned. The work also discusses the methodology of the best estimate plus uncertainty approach when applied to the safety analysis of research reactors for licensing purposes.

Key words: best estimate plus uncertainty, research reactor, RELAP5-3D, safety analysis

INTRODUCTION

IAEA safety requirements GS-R-1 recommend that prior to the granting of authorization the licensee shall submit a detailed safety analysis of the plant to the regulatory body, in accordance with clearly defined procedures. The regulatory authority needs to certify that the Safety Analysis has captured relevant postulated initiating event analysis that may pose risk to the facility, the staff of the facility, the public, and the environment [1].

Although conservative assumptions may largely be valid for designing equipment and safety analysis, they may not provide the operational and systematic insight necessary to support safe operation and maintenance, unlike the realistic approach.

An improved computing power and, also, increase of knowledge about plant phenomena, have helped in carrying out most safety analyses using best estimate (BE) system codes. The results of such analyses are accepted by the regulatory authority only if an appropriate evaluation of the uncertainty related to the results is provided. The advantage is that these codes provide more realistic information on the status of the plant, allowing direct measures or predictions of the "real" safety margins concerning uncertainty. Knowledge of the actual operational and transient conditions allows the operator to increase the performances of the plant without decreasing the safety margins. Currently, the best estimate plus uncertainty (BEPU) approach is only connected with nuclear power plants (NPP) and only with accident analysis. In the future, the same approach can be extended to the entire nuclear reactor safety process and to any nuclear installation including research reactors.

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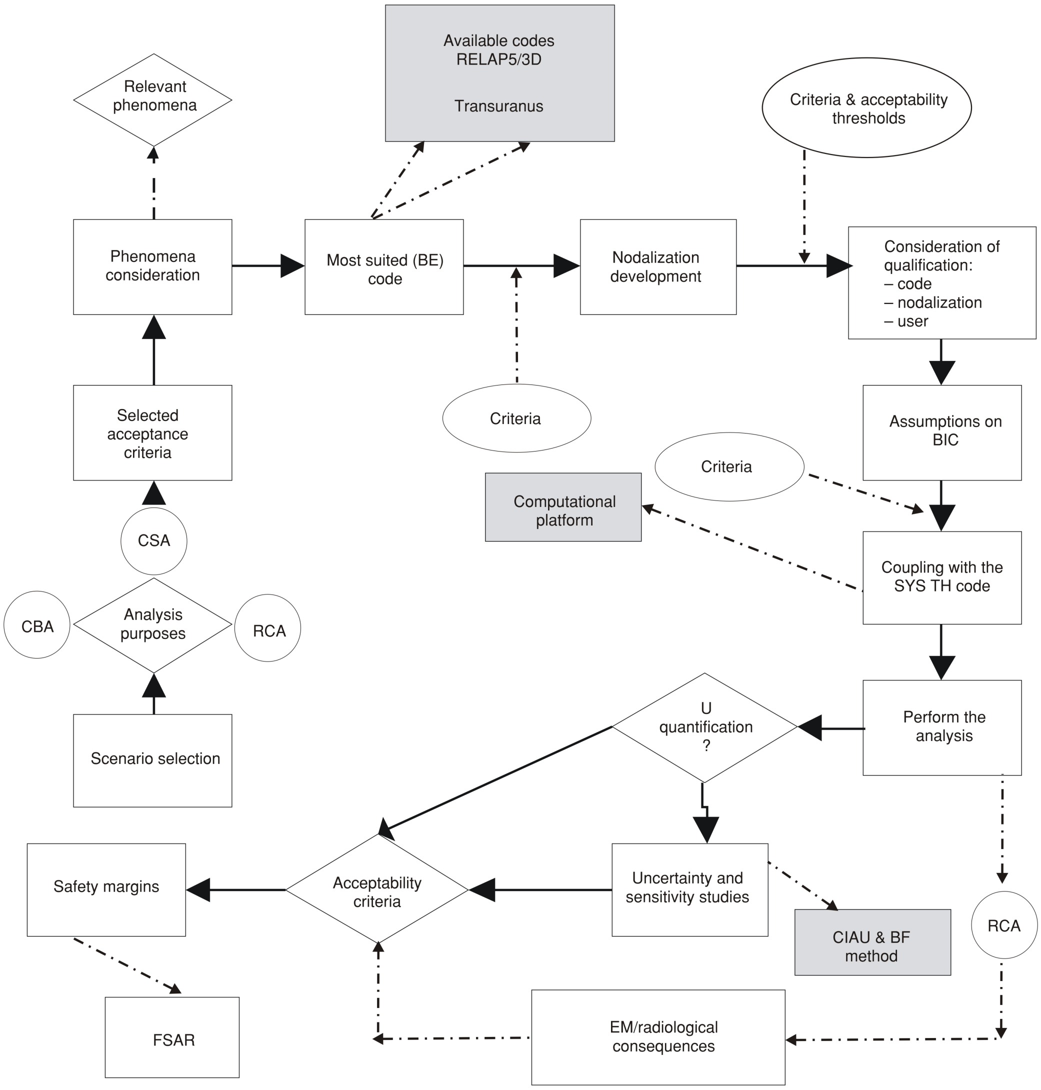

Figure 1. Flow chart of BEPU approach

BEPU APPROACH

The use of conservative tools and hypotheses in the evaluation of safety limits of nuclear reactors has limitations due to the current trend of industries to increase power production and limit economic losses resulting from uncertainties. The BEPU approach tries to address this issue, as it is based on realistic data and advanced numerical tools and methods, ensuring a direct measure of the safety margin of the plant. The same approach can be applied to research reactors when addressing safety margin issues and uncertainties in their safety analysis. The application of the BEPU approach to research reactors can be based on the existing BEPU methodology for NPP, as shown in fig. 1, without reinventing the whole process all over again.

Recently, the application of BEPU is receiving much attention in nuclear reactor safety analysis. In comparison with conservative methodologies, BEPU adopts best estimate codes and realistic input data which include uncertainties to quantify the limiting values, such as peak cladding temperature (PCT) for loss-of-coolant accidents (LOCA) [2]. It does not rely solely on the application of best estimate codes, but also on factors such as the evaluation of computer code model uncertainties, conservative assumptions on boundary and initial conditions, and the availability of system components [3]. BEPU has been identified by some organizations as being a more comprehensive method for the licensing process. Even though it has never been applied to research reactors, it can be a good approach for quantifying safety margins, as well as a preparation for emergency operational procedures (EOP) [4].

As a recent example, within the licensing process of the NPP Atucha II pressurized heavy water reactor (PHWR), the BEPU approach was selected for the issuing of Chapter 15 on final safety analysis report (FSAR). Slovak Nuclear Regulatory Authority (UJD

SR) guides for safety analysis also recommend the application of a combined or best estimate plus uncertainty approach when performing the deterministic safety analysis to support the licensing process [5, 6]. The BEPU approach has also been adopted by some vendors. Westinghouse proposed a methodology named automated statistical treatment of the uncertainty method (ASTRUM) for realistic large break LOCA (LBLOCA) analysis. AREVA implemented the GRS method to evaluate the convolution of LBLOCA uncertainty contributors to PCT [7, 8].

At the moment, due to lack of adequate knowledge to execute all the procedures involved in the use of the BEPU approach, conservative and combined approaches are more widely used.

BEPU principles

The BEPU approach does not differ significantly from the combined approach. They are similar in the selection and classification of events, acceptance criteria, availability of systems and components, operator action and user effect, presentation and evaluation of results, and quality assurances. The difference is in the use of nominal values (BE) as the initial boundary conditions and, secondly, the evaluation of the uncertainty associated with these calculations.

The BEPU approach proposed here follows current practices on deterministic accident analyses, but it also includes some key features for addressing the particular needs of the application. It credits the concept of evaluation models (EM) and comprises three separate possible modules, depending on application purposes:

– performance of safety system counter measures (EM/CSA),
– evaluation of radiological consequences (EM/RCA), and
– review of component structural design loadings (EM/CBA), where the acronyms CSA, RCA, and CBA, stand for core safety analysis, radiological consequence analysis, and component behavior analysis.

Figure 1 shows a flow chart of the BEPU approach [9].

The computational tools include:

– best estimate computer codes,
– nodalizations, including procedures for development and qualification,
– uncertainty methodology, including the procedure for qualification, and
– computational platforms for coupling and interfacing inputs and outputs from codes and nodalizations in question.

The qualification of the nodalization and coding is a vital process of the BEPU approach. A nodalization can be considered as qualified when it:

– has a geometrical consistency with the plant involved,
– reproduces the nominal measured steady-state condition of that plant, and
– shows a satisfactory behavior in time-dependent conditions, in accordance with the time-dependent data of any test performed or, if available, with any actual transient in the nuclear reactor plant.

Therefore, the qualification of the nodalization has been divided in two separate processes: steady-state and on-transient.

In principle, whenever a best estimate method is applied for licensing purposes, uncertainty quantification is required. In reality, results of code calculations do not give exact information on the behavior of a nuclear reactor during postulated accident scenarios simulation. Therefore, in order to be meaningful, best estimate predictions of plant scenarios must be supplemented by uncertainty evaluations. The code scaling, applicability and uncertainty (CSAU) evaluation methodology was developed by the United States Nuclear Regulatory Commission (USNRC), its contractors and consultants. CSAU's purpose is to address, in a unified and systematic manner, questions related to the scaling capability of a best estimate code, its applicability to scenarios of interest to NPP safety studies and the evaluation of uncertainties in calculating parameters of interest when the code is used to calculate a specified scenario and NPP design. CSAU is a systematic procedure that leads to a quantified evaluation of code calculation uncertainty [9].

The uncertainty method based on accuracy extrapolation (UMAE) is the prototype method for the consideration of the "propagation of code output errors" approach for uncertainty evaluation. The method does not focus solely on the evaluation of individual parameter uncertainties, but on the propagation of errors from a suitable database calculating the final uncertainties by extrapolating the accuracy from relevant integral experiments. To ensure the best use of the code in predicting reactor behavior, extrapolation of accuracy is calculated by finding the differences between measured and calculated quantities of the reactor.

DESCRIPTION OF THE EXPERIMENTAL FACILITY

Description of GHARR-1

The Ghana Research Reactor-1 (GHARR-1) is a 30 kW (thermal) miniature neutron source research reactor (MNSR) located at Kwabenya in Accra, Ghana. The reactor is a miniature neutron source reactor which was obtained under the Project and Supply Agreement between the International Atomic Energy Agency (IAEA), China Institute of Atomic Energy

(CIAE), and the Government of Ghana in 1994. The reactor began operation on March 15, 1995 and has since been used for neutron activation analysis, experiments and training in Nuclear Science and Technology. The reactor is fuelled with U-Al alloy with a U-235 enrichment of 90.2%. The reactor is cooled by natural convection. The thermal neutron flux density is about 1 10^{12} $cm^{-2}s^{-1}$ in the inner irradiation channels of the annular reflector. It has only one control rod. As part of the ongoing global effort to convert research reactors from highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel, the GHARR-1 fuel will be replaced by a core consisting of uranium dioxide (UO_2 clad with Zircaloy-4 alloy) fuel pins with a nominal enrichment of 12.5% and a maximum thermal power level of 34 kW to achieve the same flux density for its utilization. The main properties of GHARR-1 are shown in tab. 1 [10].

Fuel pins are arranged in 10 multi-concentric circle layers with the pitch distance of 10.95 mm. Due to the physical core design characterized by an under-moderated system, a large negative temperature feedback coefficient of reactivity is achieved. The core region of GHARR-1 is located 4.7 m under water, close to the bottom of a watertight reactor vessel. The quantity of water is 1.5 m^3 in the vessel, which serves the purpose of radiation shielding, moderation and as a primary heat transfer medium. In addition, heat can be extracted from the water in the vessel by means of a water-cooling coil located near the top of the vessel. The water-filled reactor vessel is in turn immersed in a water-filled pool of 30 m^3.

Table 1. The main parameters of GHARR-1 MNSR

Parameter	HEU	LEU
Reactor type, rated thermal power [kW]	Pool tank, 30	Pool tank, 34
U-235 enrichment [%]	90.2	12.5
Core shape	Cylindrical	Cylindrical
Core diameter [cm]	23	23
Core height [cm]	23	23
Fuel element shape	Thin rod	Thin rod
Fuel element number in the core	344	348
Fuel/cladding material	U-Al alloy/ Al alloy (Al-303-1)	UO_2/ Zircaloy-4
Fuel rod diameter [mm]	4.3	4.3
Fuel length [mm]	230.0	230.0
No. of fuel rod positions	350	350
No. of dummy elements/thickness [mm]	6/0.6	2/0.6
Dummy element material	Al	Zircaloy-4
Material of control rod guide tube	Al	Zircaloy-4

Heat removal from the core of GHARR-1

The heat generated by nuclear fission is conducted through the fuel meat to the fuel cladding, and transferred to the coolant by convection. Cold water is drawn through the inlet orifice by natural convection. The water flows past the hot fuel elements and comes out through the core outlet orifice. The hot water rises to mix with the large volume of water in the reactor vessel and to the cooling coil. Heat passes through the walls of the container to the pool water. In fig. 2 it can be seen how heat is removed from the core and the heat transfer mechanism. The core inlet flow orifice slows down the natural circulation of fluid through the core.

Description of the transient

Research reactor safety analysis calculations are simulated based on two selected broad accident categories, *i. e*., reactivity insertion accidents and loss-of-flow accidents (LOFA).

Reactivity accident analyses are carried out by systematically considering transients with and without scram events, known respectively as protected and unprotected transients or self-limited transients. Unprotected reactivity transients deal with the determination of reactivity insertion limits imposed by clad melting temperature. On the other hand, LOFA simulations have been limited to the investigation of protected transients only. With the reactor shutdown system enabled, all LOFA simulations predict that clad temperature will remain well below the clad melting point and that no flow instability takes place in the cooling channels. Data on reactor power and clad temperature responses in case of an unprotected event are still limited [12].

Transient analysis of core blockage was performed using the RELAP5-3D code, which is a best estimate code. Although the exact probability of this event is unknown, it is assumed that it will not be as high as that of the open pool configuration. Core blockage may happen as a result of fuel swelling or material falling into the core to block the flow channel. The MNSR reactor has an upward flow and, because of this, the probability of fuel swelling to block the flow channel is much higher than an object blocking the flow, which may happen as a result of some object being unintentionally left in the flow channel during maintenance. Also, since the reactor liquid flow is laminar and there are no external pumps, dragging material into the core is almost impossible.

In this study, the peak coolant channel of the GHARR-1 MNSR reactor was blocked to prevent the flow of fluid into that channel, resulting in loss of cooling at the peak. The transient was considered at maximum power of 34 kW.

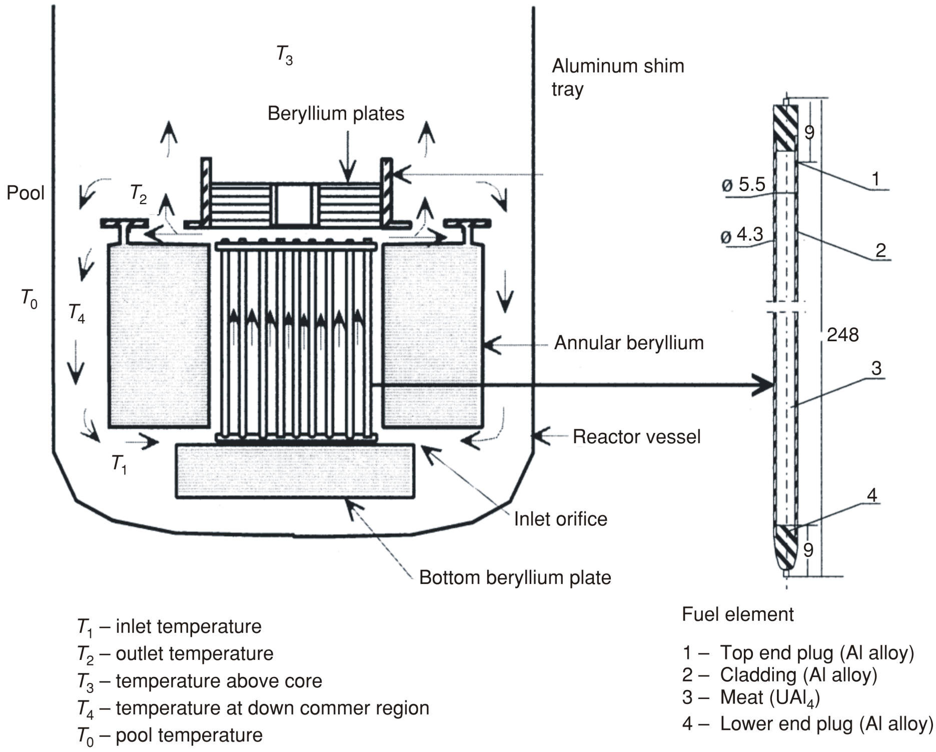

Figure 2. Schematic diagram of natural convective flow of coolant water within the reactor vessel and enlarged view of fuel element (all dimensions in mm)

CODE AND NODALIZATION

The RELAP5-3D thermal-hydraulic nodalization includes the entire MNSR system, *i. e.*, the core, the vessel and surrounding pool, as well as the reflectors. This nodalization was developed to carry out the thermal hydraulics analysis of GHARR-1 MNSR for its ongoing core conversion from HEU to LEU.

Figure 3 shows the RELAP5-3D nodalization of GHARR-1 with elevations from top (0 m)- to- bottom (–5 m) of the pool. The pool is modeled with one pipe

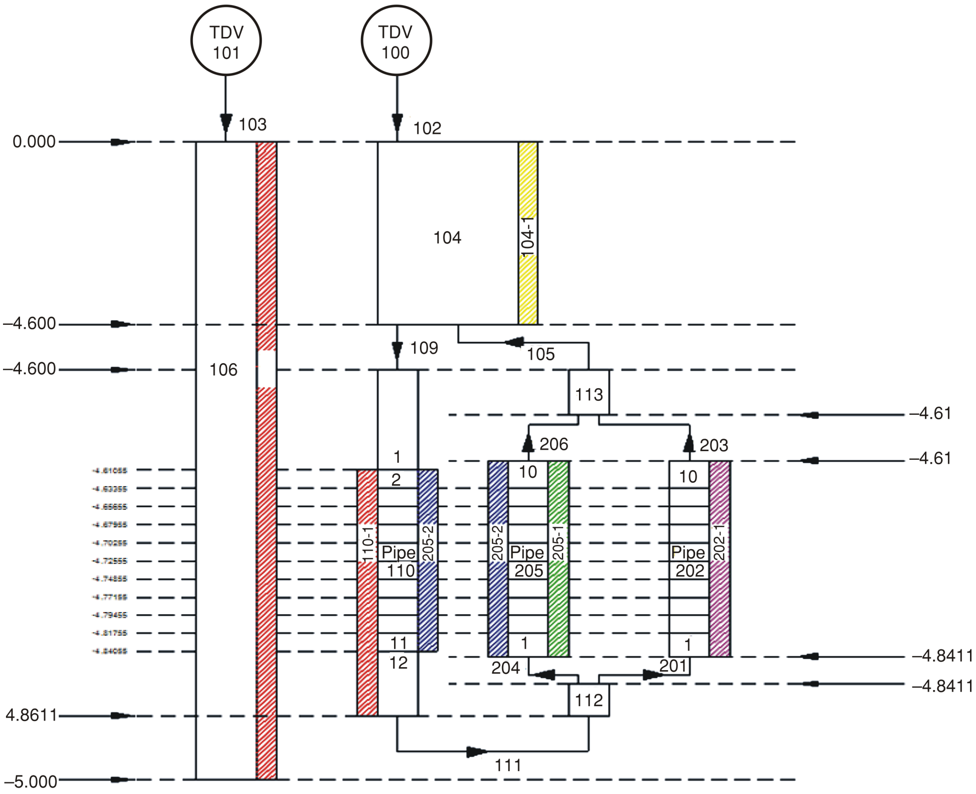

Figure 3. Nodalization of GHARR-1

(Pipe 106) and connected through heat structures (110-1) to the downcomer (Pipe 110).

The core is represented with two channels that represent the peak coolant channel (Pipe 205) and the average coolant channel (Pipe 202). In this model, the radial beryllium reflector (205-2) was modeled to transfer heat from the average pin coolant channel to the downcomer. Perfect mixing of the coolant water was assumed as it emerges from the top of the core.

Peak fuel channel flow blockage was simulated by closing the valve that connects the inlet and the peak channel.

The power distribution in the core and the reactivity feedback coefficients needed for the calculation were calculated by using the Monte Carlo N-Particle (MCNP) transport code and validated with experimental data [11].

RESULTS

The GHARR-1 reactor core was designed to accommodate sufficient natural convective flow to maintain a continuous flow of water throughout the core. The objective was to avoid significant boiling and to restrict possible steam bubbles on the surface of the fuel element.

In this study, the blockage of the peak coolant channel was considered. The reactor power for the transient was 34 kW, as shown in fig. 4. Steady-state conditions were reached after 2000 s of calculation. The blockage was simulated by completely blocking the peak coolant channel at the beginning of the transient and, as a consequence, there was no more fluid flow in this channel. During the transient, the temperature of the coolant in the core increased and reached the point of saturation in about 250 s, after which it remained constant. At that instant, boiling occurred in the peak coolant channel, as shown in figs. 5 and 6. The said phenomena caused a decrease in the mass flow and oscillations of the void in the blocked channel. Figure 7 indicates the mass flow in the peak channel.

The limiting factor for the MNSR is fuel temperature. The maximum fuel temperature should not re-

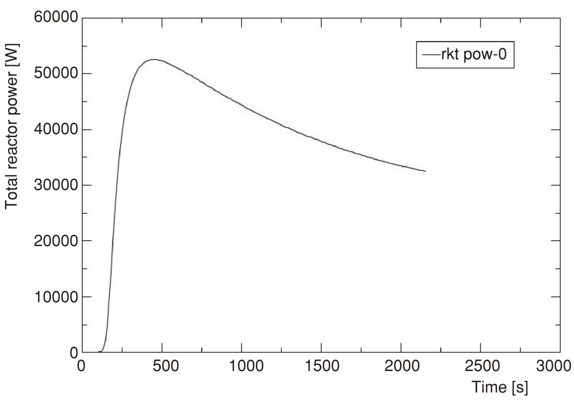

Figure 4. Power *vs.* time of transient core blockage

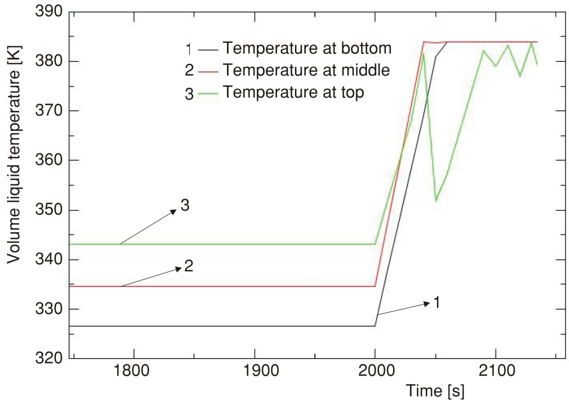

Figure 5. Fluid temperatures at different elevations of the peak coolant channel

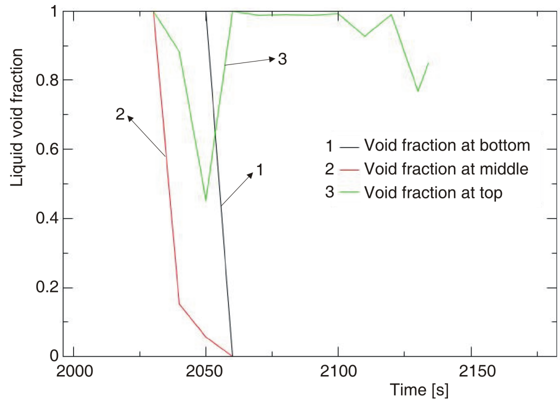

Figure 6. Void fraction of the peak coolant channel

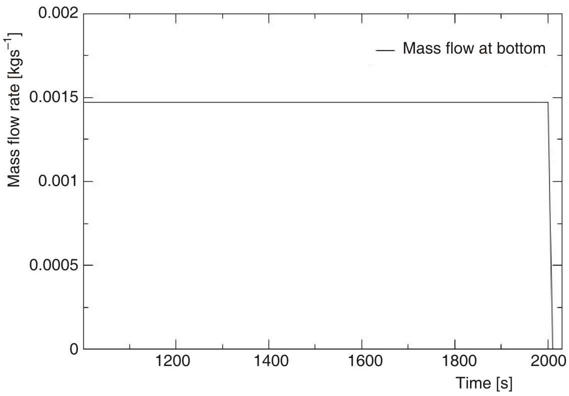

Figure 7. Mass flow rate of the peak coolant channel

sult in too much stress to the Zr-4 alloy cladding of the LEU fuel. The temperature at which melting of the Zr-4 cladding occurs is 2123.15 K. Since the reactor operates at low power, the reactor transient cannot generate a high enough temperature that leads to core meltdown. The cladding temperature at the bottom, the middle and at the top, as can be seen in fig. 8, was significantly lower than the melting temperature of the cladding, showing that the considered transient will not pose any significant threat to the core. It should be noted that the occurrence of boiling can cause the oxi-

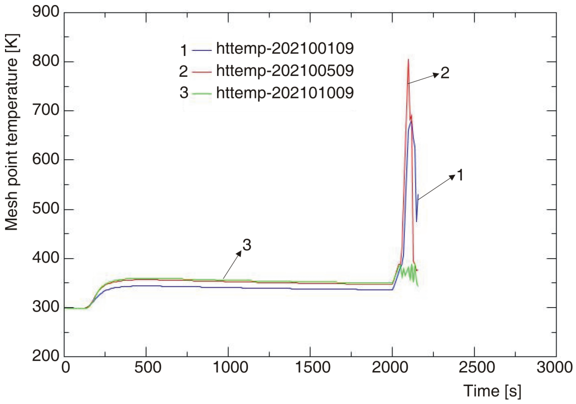

Figure 8. Clading temperatures in the peak coolant channel

dation of the cladding, but this has not been considered in this analysis.

As a safety measure, when the temperature difference between the inlet and outlet coolant of the reactor increases, the buoyancy and circulating head will increase to make the flow velocity rise and, in turn, limit the increase in power which will eventually shut down the reactor.

CONCLUSIONS

The best estimate code (RELAP5-3D) has been used to model the GHARR-1 core. The input deck was obtained from the GHARR-1 centre and prepared by the Argonne National Laboratory, based on the core conversion safety analysis from HEU to LEU.

Loss of coolant flow can lead to a weak feedback effect related to plant thermal hydraulic events. Although boiling in the reactor core occurs after the transient, the results of the GHARR-1 MNSR peak coolant channel blockage show that the temperatures are still within safety limits. The results also show that the temperatures of the cladding of the peak coolant channel's fuel are below the melting point, therefore confirming the reactor's safety for the transient.

The BEPU approach could not be fully applied due to two main reasons.

Nodalization could not be qualified because the core was modeled with just two channels; the peak and other channels may not give accurate results of the reactor's core blockage.

Insufficient data on the reactor to perform the uncertainty analysis, real data being an important part of the BEPU approach.

Thus, the plan is to improve the model in the future by coupling neutronic and thermal-hydraulic codes in order to obtain more realistic results for such types of accidents. More transient analysis needs to be done via a more qualified and more detailed nodalization that will better represent the GHARR-1 reactor. Therefore, there is a need for the development and qualification of a new nodalization before the BEPU approach can be fully applied.

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AUTHOR CONTRIBUTINS

The RELAP5 input modifications and calculations were prepared by S. Adu, with the assistance by I. Horvatovic. The analysis and discussion of the results were carried out by all the authors, including E. O. Darko and G. Emi-Reynolds. The manuscript, together with figures, was prepared by S. Adu and I. Horvatovic. All work has been supervised by F. D'Auria.

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ПРИМЕНА BEPU ПРИСТУПА У СИГУРНОСНОЈ АНАЛИЗИ ИСТРАЖИВАЧКОГ РЕАКТОРА

За изградњу и рад нуклеарног истраживачког реактора потребно је добити одобрење од одговарајућег регулаторног тела. Један од задатака овог тела је провера да ли спроведене сигурносне анализе испуњавају сигурносне услове. Процена усклађености са сигурносним условима рађена је употребом детерминистичког приступа уз конзервативне претпоставке. Овим се осигуравају довољне сигурносне маргине у односу на лиценцне услове.

Конзервативне претпоставке представљене су у оквиру сигурносних анализа како би се компензовале несигурности повезане с недостатком знања. Увођењем рачунарских кодова најбоље процене, почиње употреба сигурносних анализа које се темеље на овом приступу. Резултати оваквих анализа могу бити прихваћени од стране регулаторног тела само ако је спроведена одговарајућа процена несигурности. Рачунарски кодови најбоље процене у стању су да прикажу реалније стање постројења, чиме се омогућава предвиђање стварних сигурносних маргина. BEPU приступ (приступ најбоље процене са несигурношћу) показао се поузданим дајући реалне резултате, ако су пажљиво испуњени сви услови. Овај рад, представља концепт BEPU приступа и начин на који се он може применити на сигурносне анализе истраживачких реактора.

Циљ рада је прорачун транзијената губитка тока при „блокади језгра” у истраживачком реактору са минијатурним неутронским извором, применом BEPU методологије. Резултати прорачуна показали су да су температуре у језгри унутар сигурносних граница и да не представљају значајну претњу за реактор у погледу топљења кошуљице. Такође, у раду се разматра методологија BEPU приступа и његова примена на истраживачке реакторе за потребе лиценцирања.

Кључне речи: најбоља процена са несигурношћу, истраживачки реактор, RELAP5-3D, сигурносна анализа