I. Ovdiienko[1], M. Ieremenko[1], Y. Bilodid[1],
Jelena Krhounkova[2]

[1] State Scientific and Technical Center for Nuclear and
Radiation Safety, Kyiv, Ukraine
[2] ÚJV Řež, a. s., Czech Republic

Experience and perspective of best-estimate approach application for RIA analysis

The use of best-estimate approach for WWER safety analysis in RIA is considered. The relevance of this problem is concerned with small margin to acceptance criteria under the conservative approach and becomes stronger under power uprate of nuclear power plants. Previous experience in this area for WWER-1000 reactor types is overviewed. The necessity to extend these activities for successful implementation of the best-estimate approach is noticed and areas of further work are discussed.

K e y w o r d s: WWER, best estimate, uncertainty analysis, reactivity-initiated accident

Ю. М. Овдієнко, М. Л. Єременко, Є. І. Білодід, Е.Крхоункова

Досвід і перспективи застосування підходів найкращої оцінки для аналізу реактивнісних аварій

Розглянуто питання використання та реалізації підходів найкращої оцінки для аналізу безпеки ВВЕР у реактивнісних аваріях. Актуальність проблеми пов'язана з малими запасами до критеріїв прийнятності при реалізації консервативного підходу, що особливо посилюється в умовах підвищення номінального рівня потужності реакторної установки. Представлено короткий огляд попереднього досвіду в цій галузі для реакторів ВВЕР-1000. Зазначено необхідність розширення робіт для успішної реалізації підходів найкращої оцінки, обговорено напрями подальшої діяльності.

К л ю ч о в і с л о в а: ВВЕР, найкраща оцінка, аналіз невизначеності, реактивністні аварії.

At present time, WWER-1000 operating companies faced the problem of small margin to acceptance criteria under implementation of the conservative approach. Regarding Ukraine, the problem is particularly significant in view of power uprate of nuclear power plants. Such situation requires introduction of the best-estimate plus uncertainty (BEPU) approach. For some accidents, such as loss of coolant accident (LOCA), the best-estimate approach is more or less developed and settled. However, for reactivity initiated accident (RIA) analysis, application of the best-estimate method could be problematic.

Regulatory documents (both in Ukraine and Czech Republic for example) define a nomenclature of neutronic calculations and so called "framed safety parameters" which should be used as boundary conditions for all WWER-1000 reactors in RIA analysis.

The best-estimate computer codes combined with conservative initial and boundary conditions (combined analysis) are used for design basis accident (DBA) analysis in RIA in the framework of safety analysis report (SAR) in Ukraine. For a given purpose, the approach is developed to include all RIA significant conservative initial and boundary conditions into a realistic model of the reactor core. The conservative values of parameters such as:

– reactivity coefficients,
– efficiency of control rod (CR) and scram weight,
– characteristics of the most loaded fuel pin, and
– thermal hydraulic characteristics

are introduced into the developed models for DBA analysis. Depending on used neutron kinetics, the approaches slightly differ but are very similar in general. Such an approach complies with IAEA recommendations.

The range of conservatism is defined by the Ukrainian regulation "Fuel Handling. Refueling in WWER-1000 Reactor. Nomenclature of Operational Neutronic Calculations and Experiments" (Energoatom, 2013), SOU NAEK 064:2013 [1]. The so-called frame safety parameters are defined. Frame safety parameters are the same for all WWER-1000 (V320+TVSA). There are slight differences only for V302/V338 designs and for fuel loadings with TVS-W (Westinghouse assemblies).

A similar table for the frame safety parameters is defined by Czech regulations as well.

As is seen from the table 1, the frame safety parameters have a wide range of changes. Moreover, the use of limit values in this range could lead to too conservative results.

Another problem is introduction of conservative assumption in into the model of best-estimate computer codes. The current approach applied for DBA in the framework of SAR for most Ukrainian NPPs is presented on example of initial event with CR ejection. This approach [2] assumes the following choice of conservative initial and boundary conditions with use of DYN3D [3] for accident analysis:

– the conservative values of initial reactor power, coolant flow rate, pressure, scram actuation setpoints etc. are defined based on operational limits, errors of their definition and development of transient under the worst scenario;

– the conservative values of reactivity coefficients are achieved with help of cross-section parameterization correction ($\nu\Sigma_f^{th}$) in the range of accuracy of its definition. *For the considered mode, cluster ejection for the state corresponding to the beginning of fuel cycle with real values of reactivity coefficients for coolant temperature, coolant density and fuel temperature $\alpha_{Tm} = -33 \cdot 10^{-3} 1/°C$; $\alpha_\gamma = +15\%/(g/cm^3)$ and $\alpha_{Tf} = -2{,}7 \cdot 10^{-3} \%/(°C)$, with appropriate correction, were received $\alpha_{Tm} = -18{,}0 \cdot 10^{-3} 1/°C$, $\alpha_\gamma = +0\%/(g/cm^3)$ and $\alpha_T = -1{,}7 \cdot 10^{-3} \%/(°C)$;*

Table 1. Frame safety parameters

Parameter	Reactor power	Moment of campaign	Frame values	
			Min	**Max**
Reactivity coefficients for fuel temperature, $\delta r/\delta T_U$, $\cdot 10^{-5}$ 1/°C	HFP	BOC-EOC	-3,2	-1,7
	HZP	BOC-EOC	—(-3,8)*	-2,2
Reactivity coefficients for coolant temperature, $\delta r/\delta T_m$, 10^{-5} 1/°C	HFP	BOC	-45,6	-18,0
		EOC	-84,0	-42,5
	HZP	BOC	—(-17,8)	2,2
		EOC	-45,1	—(-17,0)
Reactivity coefficients for coolant density, $\delta r/\delta g$, 10^{-2} 1/(g/cm^3)	HFP	BOC	0,0	—(31,0)
		EOC	—(23,7)	37,0
	HZP	BOC	0,0	—(31,0)
		EOC	—(14,3)	31,0
Reactivity coefficients for boron concentration, $\delta r/\delta C_b$, %/(g/kg)	HZP-HFP	BOC-EOC	-2,4	-1,0
		BOC-EOC		
Effective fraction of delayed neutrons β_{eff}, %	HZP-HFP	BOC	0,56	0,80
		EOC	0,50	—(0,66)
Effective prompt neutron lifetime, $l_{pn} \cdot 10^{-6}$, sec	HZP- HFP	BOC-EOC	15	37
Efficiency of working group of CR, %	HFP	BOC-EOC	—(0,48)	1,15
	HZP	BOC-EOC	—(0,48)	1,32
Scram efficiency, %	HFP	BOC-EOC	5,0	—
	HZP	BOC-EOC	3,0	—
Efficiency of ejected CR, %	HFP	BOC-EOC	—	0,30
	HZP	BOC-EOC	—	0,75

*—(-3.8) and further denote — frame value isn't set up, but value in brackets (-3.8) is used for safety analysis.

– the conservative effective β_{eff} and speed values of delayed neutrons are given for fuel burnup, at which the minimal value β_{eff} is observed (*for existing fuel cycles, the conservative value $\beta_{eff} = 0,005$ is observed at the maximal burnup 44,0 MW·days/kgU*);
– the conservative value of ejected cluster efficiency is provided by complete inserting of CR up to the core bottom, and also spatial deformation of neutron flux distribution with correction of concentration of xenon nuclei in the area of ejected cluster (*for the case considered below using the described approach, the ejected cluster efficiency is increased up to 0,30 %*);
– the conservative values of fuel pin power are provided by introducing the "hot channel" with a limiting axial profile of power distribution (first profile with a maximum in the bottom part, second at the center and third with a maximum in the top part of reactor core);
– the relative power of the most loaded pin amounts to $k_r^{cons} = 1,74$ and is defined by the maximum allowable power peaking factor ($k_r^{lim} = 1,5$) taking into account engineering factor 1,16;
– for the most loaded pin, the hot channels are modeled with maximal and minimal gas gap width;

– the minimum scram efficiency is provided taking into account an error of definition (5 %). Such efficiency is achieved by jamming of some clusters. One of the jammed clusters is located nearby fuel assembly (FA) with the most loaded pin. The fall time of scram control rods is accepted equal to the greatest design value amounting to 4 sec.

As a result of the assumed choice of conservative initial and boundary conditions, the narrow margin to acceptance criteria was obtained with regard to key safety parameters — maximal fuel and cladding temperature (Fig. 1, Fig. 2).

Problem of the narrow margin to acceptance criteria becomes stronger with an intention to increase rated reactor power, which leads to further decrease of margins to acceptance criteria.

Calculation capabilities for BEPU. A set of multipurpose neutron kinetic codes is necessary for implementation of the best-estimate approach for WWER safety analysis in RIA.

First of all, a code for preparation of a few-group cross-section library (or an existing cross-section library) is required. For this purpose, SSTC NRS uses a few-group cross-section library prepared with the HELIOS code. The SCALE code can
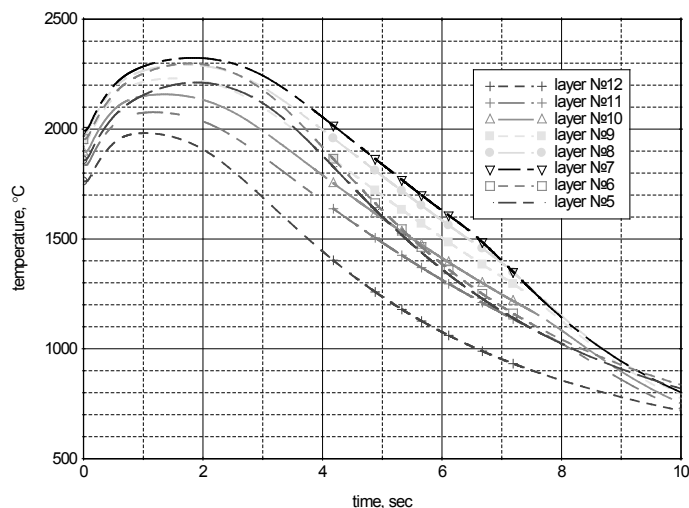
Fig. 1. Fuel temperature in the most loaded fuel pin

Fig. 2. Cladding temperature in the most loaded fuel pin

also be used for uncertainty and sensitivity analysis of cross-sections with use of nuclear data uncertainties.

At the next stage, a neutron kinetic code for steady state and transient calculation is required. For this purpose, the DYN3D code is a perfect calculation tool. Its advantage is the established approaches for change of initial and boundary conditions as uncertainty parameters in the model used previously for conservative assessment.

If it is necessary to take into account response of the secondary circuit, a coupled thermal hydraulic system code with neutron kinetics should be used. In using the GRS SUSA approach, the DYN3D/ATHLET coupling is the optimal choice. Nevertheless, the use of codes such as RELAP and TRACE is quite acceptable.

An important element of the required calculation capabilities is flexibility of models that should allow a variation of uncertainty parameters.

Besides the instrument for uncertainty and sensitivity analysis such as SUSA from GRS, as it was mentioned above, an additional code is necessary for uncertainty and sensitivity analysis of cross sections with use of nuclear data uncertainties (XSUSA also from GRS).

Previous experience and further activities. Previous experience in this issue was described by Jan Hádek, ÚJV Řež, a.s., in the report "Selected Safety and Best-Estimate Analyses of NPP with WWER-1000" on AER Working Group D Meeting on WWER Reactor Safety Analysis [4]. The results of best-estimate analysis of CR ejection with use of the GRS methodology for uncertainty and sensitivity analysis SUSA were presented. The reactor dynamic code DYN3D 3.2 was used for analysis. In the presented approach, important uncertainty parameters were taken into account such as reactivity coefficients and gas gap conductivity. But some important factors for the accident such as efficiency of ejected cluster and power axial profile were missing.

A similar approach was presented by ÚJV Řež for best-estimate analysis of the accident related with steam line break [5] in the framework of DBA analysis of SAR at the Scientific and Technical Conference "Safety Assurance of NPP with WWER".

However, to accomplish all efforts on the use of best-estimate approach for WWER safety analysis in RIA, the started
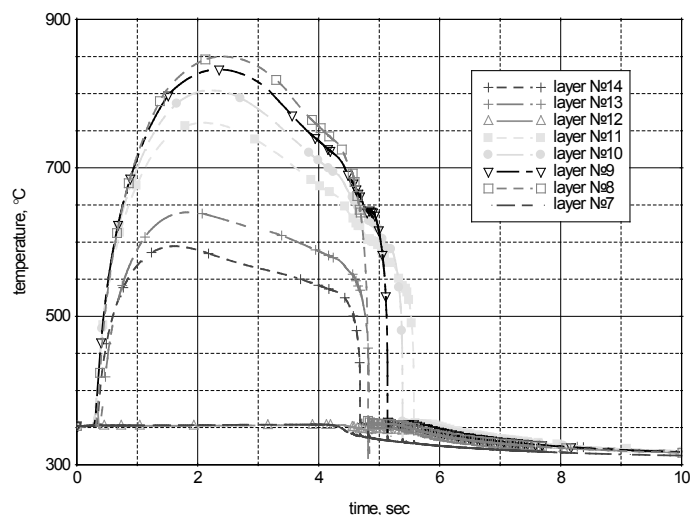
activities should be extended. For this purpose, the following steps should be taken:

– choice of significant uncertainty parameters for one of the representative RIA (ejection of CR for example). Most probably, the list of uncertainty parameters should be based on the above-mentioned table of frame safety parameters;

– variation of chosen uncertainty parameters in the computer model (reactivity coefficients, efficiency of CR and scram weight, characteristics of the most loaded fuel pin, thermal hydraulic characteristics etc.);

– performance of calculations (a great amount of cases);

– sensitivity analysis with the aim of rejecting unimportant uncertainty parameters for further safety analysis.

As a result, the elaborated recommendations for uncertainty analysis in computer models concerning safety analysis in RIA could be very useful both for the SAR developer and regulator.

Conclusions

The development of best-estimate approaches with uncertainty analysis and their implementation for WWER safety analysis in RIA are highly relevant. It is determined by a wide range of frame safety parameters for the SAR developer to cover all operational modes and the intention to increase rated reactor power.

There is ÚJV Řež experience on best-estimate analyses of NPPs with WWER-1000, but it should be extended for RIA analysis in the framework of SAR. The elaborated recommendations for introduction of uncertainty analysis into computer models for safety analysis in RIA could be very useful both for the SAR developer and regulator.

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