A. G. Samokhin and *M. Zilly*

Application of 3D coupled code ATHLET-QUABOX/CUBBOX for RBMK-1000 transients after graphite block modernization

This work describes the application and the results of transient calculations for the RBMK-1000 with the coupled code system ATHLET 2.2A-QUABOX/CUBBOX which was developed in GRS. Within these studies the planned modernization of the graphite blocks of the RBMK-1000 reactor is taken into account. During the long-term operation of the uranium-graphite reactors RBMK-1000, a change of physical and mechanical properties of the reactor graphite blocks is observed due to the impact of radiation and temperature effects. These have led to a deformation of the reactor graphite columns and, as a result, a deformation of the control and protection system (CPS) and of fuel channels. Potentially, this deformation can lead to problems affecting the smooth movement of the control rods in the CPS channels and problems during the loading and unloading of fuel assemblies. The present paper analyzes two reactivity insertion transients, each taking into account three graphite removal scenarios. The presented work is directly connected with the modernization program of the RBMK-1000 reactors and has an important contribution to the assessment of the safety-relevant parameters after the modification of the core graphite blocks.

Anwendung des gekoppelten 3D-Code-Systems ATHLET-QUABOX/CUBBOX auf Transienten im RBMK-1000 nach der Modernisierung der Graphit-Blöcke. *Der vorliegende Artikel beschreibt die Anwendung des gekoppelten Code-Systems ATHLET 2.2A-QUABOX/CUBBOX der GRS auf Transienten im RBMK-1000 Reaktor und deren Ergebnisse. Dabei wurde die geplante Modernisierung der Graphit-Blöcke im RBMK-1000 untersucht. Während des Langzeitbetriebs des Uran-Graphit-Reaktors RBMK-1000 wird eine Änderung der physikalischen und mechanischen Eigenschaften der Graphit-Blöcke im Reaktor beobachtet. Diese Änderungen sind eine Folge von Strahlungs- und Temperatureinwirkungen. Sie haben zu Deformationen der Graphit-Säulen und dadurch zu Verformungen des Regelungs- und Schutzsystems (CPS) und der Brennstoffkanäle geführt. Potentiell können diese Verformungen Probleme in der gleichmäßigen Bewegung der Steuerstäbe in den CPS-Kanälen sowie beim Be- und Entladen von Brennelementen verursachen. Der vorliegende Artikel analysiert zwei Reaktivitätstransienten, wobei jeweils drei unterschiedliche Mengen Graphit entfernt wurden. Die vorgestellte Arbeit ist direkt verbunden mit der Modernisierung der RBMK-1000-Reaktoren und liefert einen wichtigen Beitrag zur Bewertung sicherheitsrelevanter Parameter nach der Modernisierung der Graphit-Blöcke.*

1 Introduction

In 2006, the cracking of graphite blocks in a RBMK-1000 reactor has been observed for the first time. Graphite block deformations were detected at Units 1 and 2 of Leningrad Nuclear Power Plant (NPP). In 2012, a graphite recovery procedure was performed on Unit 2 of Leningrad NPP. The main goal of this procedure was to cut and remove parts of the graphite blocks. The total graphite removal for the whole reactor core after the recovery procedures was about 3 % by volume.

According to the Russian regulatory laws, during the NPP's safety evaluation, a range of design basis accidents calculations must be performed by the utility. The list of initializing events to be considered is described in Russian safety guides. Apart from the utility, the technical support organization (SEC NRS) of the Russian regulatory authority (Rostechnadzor) performs the same calculations, but using a different code system and independent modelling. After comparing the results, Rostechnadzor takes a decision about whether or not the NPP can continue operation.

Due to the huge size of the RBMK-1000 reactor core, the analysis of reactivity transients requires special attention and specific methods. Such transients include, for example, the withdrawal or fallout of control and protection system (CPS) rods, or the reduction of the feed water flow which causes changes of the vapor reactivity coefficient. The analysis of the aforementioned processes requires coupled neutronics and thermo-hydraulics codes, the use of point kinetics is not suitable. QUABOX/CUBBOX [1] and ATHLET 2.2A [2], developed in GRS, are an example of coupled codes which SEC NRS uses for RBMK-1000 transients analysis. The coupled code-systems ATHLET-QUABOX/CUBBOX can correctly determine reactivity changes during transients in the large RBMK-1000 reactor core.

This article examines the influence of the graphite removal on CPS withdrawal transients and presents corresponding simulation results. The models applied are based on Unit 1 of the Kursk NPP. In spite of the fact that until now not more than 3 % of graphite has been removed in RBMK-1000 NPPs, it is assumed that this amount will have to be increased in the future to 6 %-12 %, depending on the NPP. Therefore, in this article, reactivity transients are considered for the reference case without graphite removal are compared to those with 6 % and 12 % graphite removal.

2 Kursk NPP Unit 1 Model

2.1 *QUABOX/CUBBOX core model*

QUABOX/CUBBOX is a neutronics code which was developed for detailed 3D reactor core modeling. It solves the two-group diffusion equation taking into account fuel temperature and coolant density feedbacks [1]. Additionally for RMBK also moderator temperature feedbacks are considered.

The QUABOX/CUBBOX RBMK core model consists of 2488 fuel assembly equivalent positions: 604 reflector fuel assembly equivalent positons and 1884 fuel assembly positions, which include additional absorbers, control rods and water columns. Axially, each equivalent fuel assembly is divided into 30 nodes. As input parameters, QUABOX/CUBBOX requires the following plant data: The core loading map, the coolant flow rate distribution, positions of control rods, burn-up distribution, inlet coolant temperatures. Cross-section (XS)-libraries for RBMK-1000 core calculations were prepared by SEC NRS for the reference case and the cases with 6% and 12% graphite removal. The libraries provide the two-group D_1, D_2, Σ_{a1}, Σ_{a2}, Σ_R, $\nu\Sigma_{f1}$, $\nu\Sigma_{f2}$ (see Section 7 for the nomenclature), as 5D tables with fuel and graphite temperature, coolant density, burn-up and xenon concentration as independent parameters. The core loading of Kursk NPP Unit 1, which is shown on Fig. 1, is similar to the loading pattern considered in [3].

2.2 *ATHLET model*

ATHLET is an advanced best-estimate thermal-hydraulics code which has been developed for the simulation of design basis and beyond design basis accidents (without core degradation) in light water reactors, including RBMK reactors [2].

The ATHLET model of the RBMK-1000 developed at SEC NRS and GRS represents the main coolant circuit (MCC), the feed water system, the steam lines system, main safety valves. The model is similar to the one used in [3] and depicted in Fig. 2.

In the calculation, explicit models were considered for the heat exchange between fuel, cladding and coolant as well as between graphite moderator blocks and channel tubes with the corresponding fuel channel (FC). Heat transfer structures in the water communications and the steam-water communications were also represented.

The core is represented by equivalent fuel channels, each of them consisting of three parts: a core inlet part connected to a pipeline for water communications and control valves, an active core channel part, and a core outlet part. The core outlet part connects to a pipeline simulating steam-water communications. In the ATHLET nodalization scheme (see Fig. 2), 10 types of FCs with different power are considered in the left and right halves, respectively. This 20-channels model was chosen as a compromise between running time and calculation accuracy. It is assumed that all channels in one group have the same thermohydraulic parameters and the division into groups was made according to the channels' power.

3 Initial and boundary conditions

As initial conditions for the transient simulations, nominal power stationary conditions are assumed. The reactor power is at nominal level. Pressure and drum-separator (DS) level regulators are actuating. Automatic power control rods (ACR), which are actuated by in-core detectors, are in operation.

The initial conditions are characterized by:

- 3 main circulation pumps (MCPs) in operation;
- 4 feed water pumps (FWPs) in operation;
- the water reserve in deaerators is 480 tons;
- the pressure in drum separators (DSs) is ~6.87 MPa;
- the pressure controller for DS is in operation and can actuate the turbine generator (TG) stop control valve (TSCV);
- the steam flow rate for station supply is 200 t/h;
- the feed water temperature is 160 °C;
- the water levels in the DSs are kept up by the level controllers at the position 350 ± 50 mm.

The main technological parameters of Kursk NPP Unit 1 are presented in Table 1. The calculation process of the ATHLET-QUABOX/CUBBOX coupled code system consists of three phases: "steady state" – the model initialization, "zero

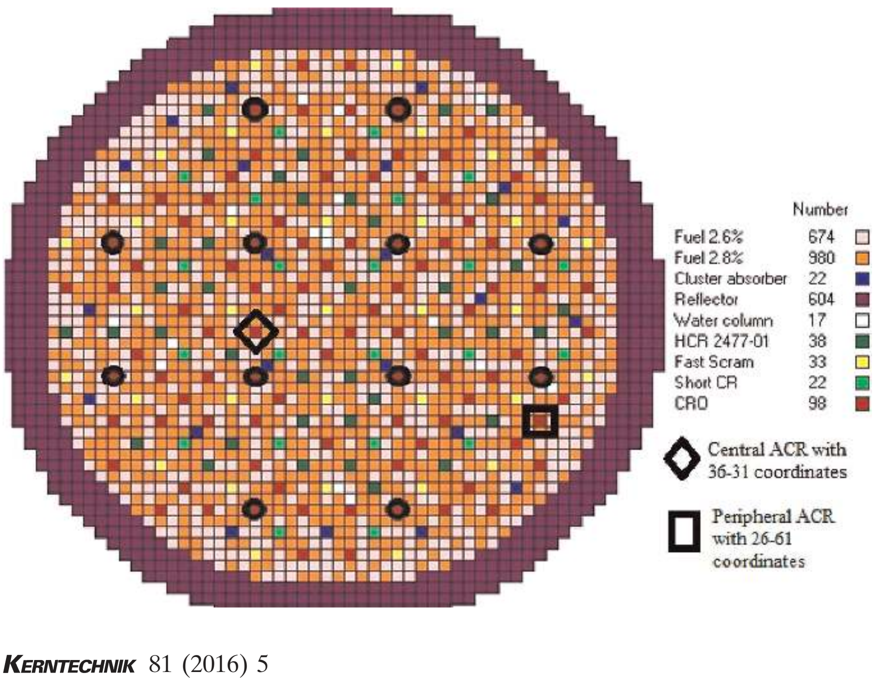

Fig. 1. Kursk NPP Unit 1 core loading (the 12 local automatic control rods (LACR) radial positions are indicated by black circles)

transient" – the stabilization of reactor parameters and "transient" – the calculation of a transient. In the present model, the stabilization of the reactor parameters was achieved after 1 850 s for the withdrawal transients. The transient calculations start after these stabilization periods.

4 Modeling results

The following transients were modeled in this work:

- Central CPS rod withdrawal (coordinates 36–31);
- Peripheral CPS rod withdrawal (coordinates 26–61);

It should be mentioned here that these transients must be performed during the RBMK-1000 NPP's safety justification according to the Russian safety guides.

4.1 Central CPS rod withdrawal

After the zero transient simulations, the reactor parameters are at nominal values, the fuel temperature in the channels does not exceed 1 500 °C, the fuel cladding temperature in the channels does not exceed 327 °C, the reactor heat capacities are at nominal level.

The transient simulation starts at 1 850 s. The central CPS rod with coordinates 36–31 and 700 cm initial insertion is withdrawn with a velocity of 40 cm/sec. For the analyses of CPS rods withdrawal, the most conservative situation was considered, i. e. when the initializing event is a manual rod withdrawal by the operator. This is done by intermittently switching on the CPS drive many times. During this withdrawal, the operator's mistake cannot be detected by the controlling scheme, because the penalty command with confirmation is formed. Also, the half rod withdrawal blocking system is not active in this case.

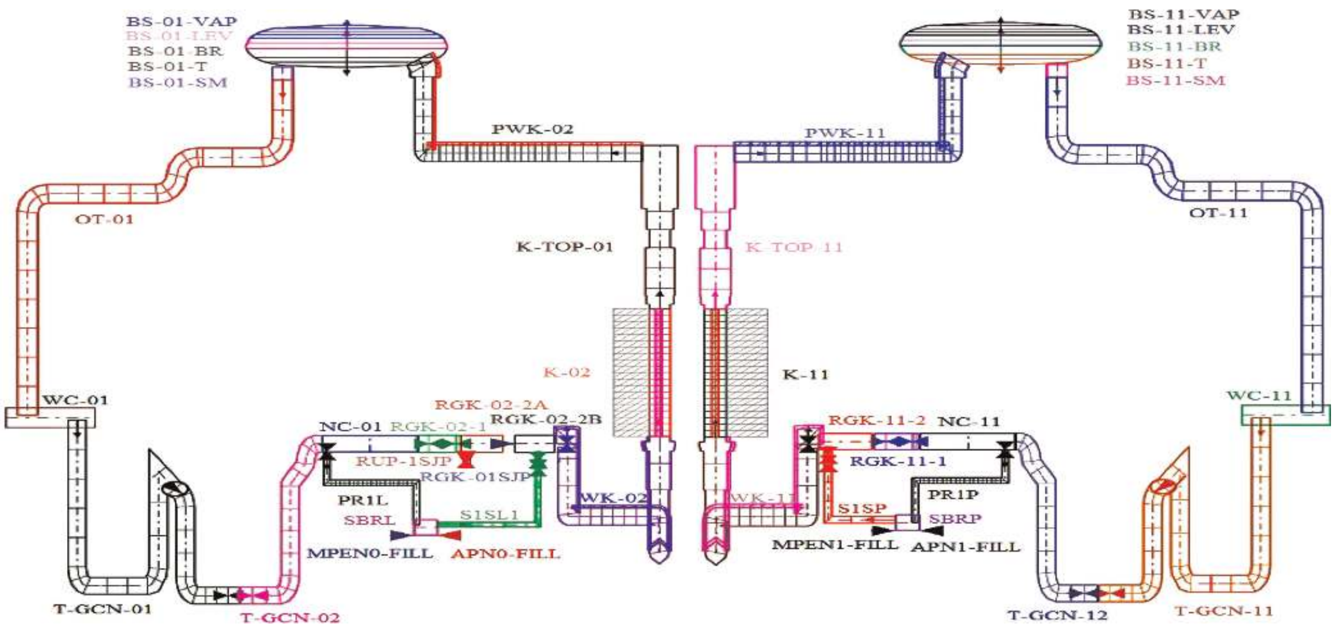

Fig. 2. RBMK-1000 MCC ATHLET model

Table 1. Main technological reactor parameters before measurements

No.	Parameter	Value		
		No graphite removal	6 % graphite removal	12 % graphite removal
1	Thermal power [MWt]	3 142	3 111	3 097
3	Mean burnup (E_{av}) [MW day/tU]	1 602	1 602	1 602
5	MCC flow rate in left/right half of MCC [m^3/h]	23 535/23 520	22 003/21 800	21 901/21 723
6	Feed water flow rate (G_{fw}) in left/right half of MCC [t/h]	2 221/2 436	2 702/2 703	2 674/2 717
7	Mean DS pressure in left/right half of MCC [MPa]	6.88/6.86	6.87/6.86	6.87/6.87
8	Mean header pressure (P_{pH}) in left/right half of MCC [MPa]	8.25/8.16	7.68/7.69	7.69/7.71
9	Mean suction header (SH) temperature in left/right half of MCC [°C]	266.3/265.7	267.8/267.7	267.9/267.5
10	Mean feed water temperature in left/right half of MCC [°C]	161.7/162.3	161.2/161.1	161.1/161.1
11	Radial power peaking factor (K_r) [–]	1.41	1.38	1.29

As an additional assumption, the failure of the detector which forms the first setpoints deviation signal was considered. The failure of the second detector was also assumed, which is a part of the signal detecting flux increase. The failure of the withdrawal blocking system after warning signals was chosen as "non-controlling failure" during the NPP operation. Graphics of the fuel temperature, fuel rods cladding temperature, maximum power of the hot channel, reactivity and total reactor power during the withdrawal for the three graphite removal cases are presented in Figs. 3–8.

As can be seen from Figs. 3–8, the maximum fuel temperature in the hot channel increases by about 150–200 °C during the central CPS rod withdrawal for all three graphite removal cases. It does not exceed 1750 °C. The fuel rods cladding temperature increases by 3–5 °C and does not exceed 331 °C. The local power in the hot channel increases by 0.4–0.6 MWt, stabilizes and does not exceed 3.0 MWt, which is the operational limit. The total reactor power increases approximately by 90 MWt, and then stabilizes as the ACR start working.

The central CPS rod withdrawal is thus compensated by a group of ACR, and the increase of the local power in the fuel channels and the total reactor power is partly suppressed. Moreover, the results obtained for the three different graphite removal cases are very close to each other, such that there are no additional procedures required to control this transient for all graphite removal cases.

4.2 Peripheral CPS rod withdrawal

The peripheral CPS rod with the coordinates 26–61 was withdrawn in this transient. The initial insertion depth 290 cm was chosen for this transient. The initial conditions are the same as in previous section for the central CPS rod removal. Graphics of the fuel temperature, fuel rods cladding temperature, maximum power of the hot channel, reactivity and total reactor power during the withdrawal for all three graphite removal cases are presented in Figs. 9–14.

As can be seen from Figs. 9–14, the maximum fuel temperature in the hot channel increases maximally by 150 °C and does not exceed 1750 °C in all three graphite removal cases, during and after the removal of the peripheral CPS rod. The fuel rods cladding temperature increases maximally by 10 °C and does not exceed 330 °C. The local power in the hot channel increases maximally by 0.3 MWt, then stabilizes and does not exceed 2.9 MWt. The total reactor power in-

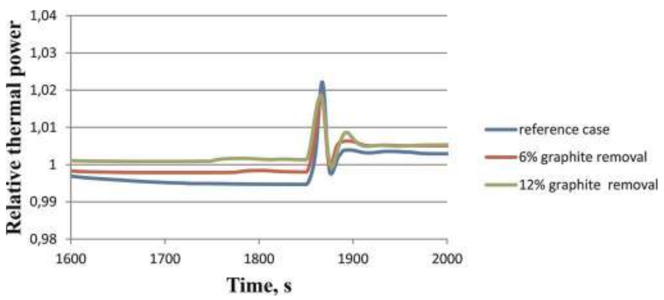

Fig. 3. History of the total reactor power during the withdrawal

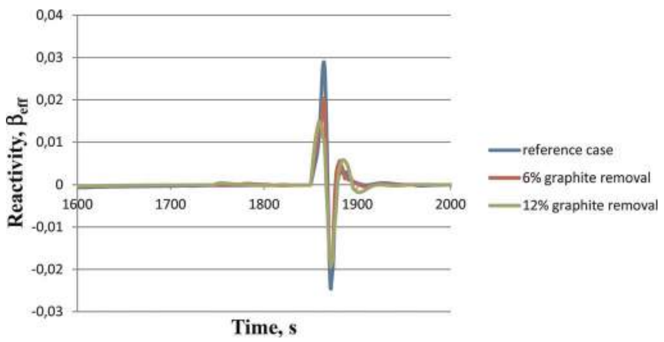

Fig. 4. Reactivity evolution during the withdrawal

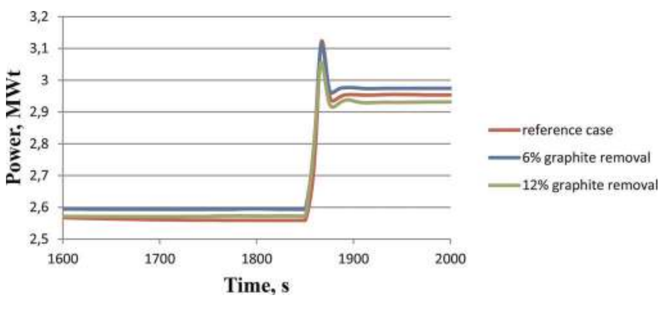

Fig. 5. Maximum power history of the hot channel

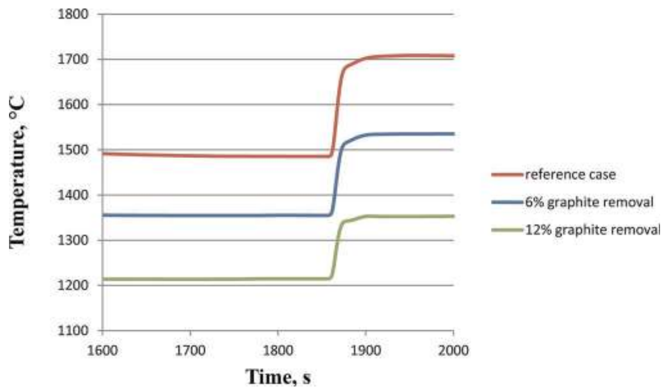

Fig. 6. Fuel temperature history of the hot channel

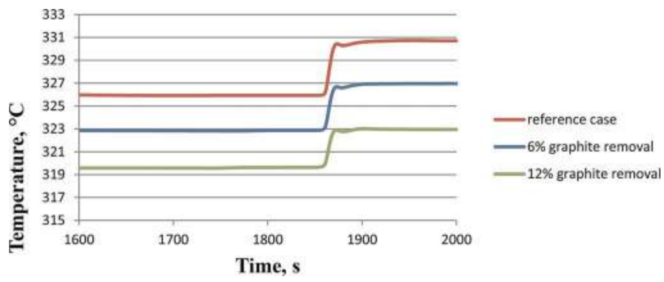

Fig. 7. Assembly cladding temperature history of the hot channel

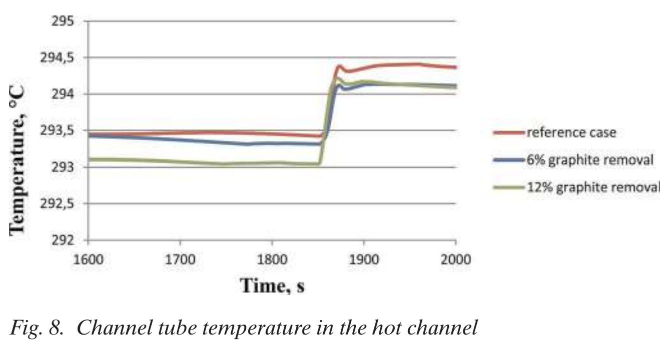

Fig. 8. Channel tube temperature in the hot channel

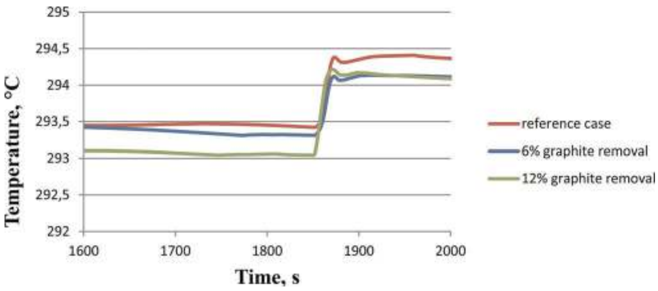

Fig. 9. History of the total reactor power during the withdrawal

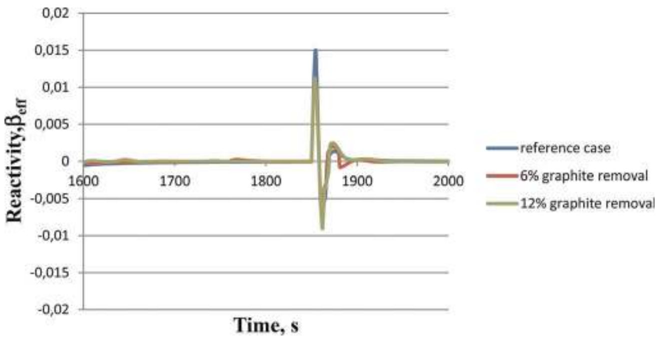

Fig. 10. Reactivity evolution during the withdrawal

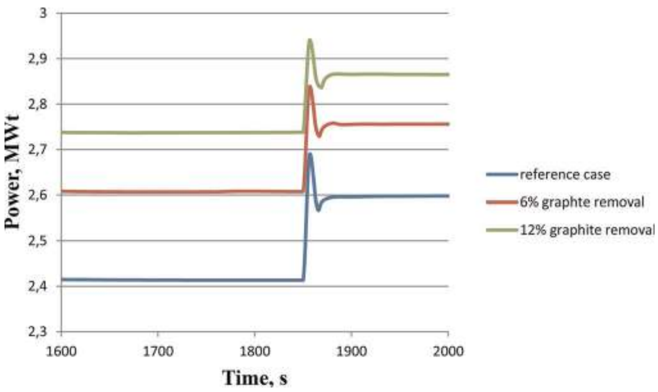

Fig. 11. Maximum power history of the hot channel

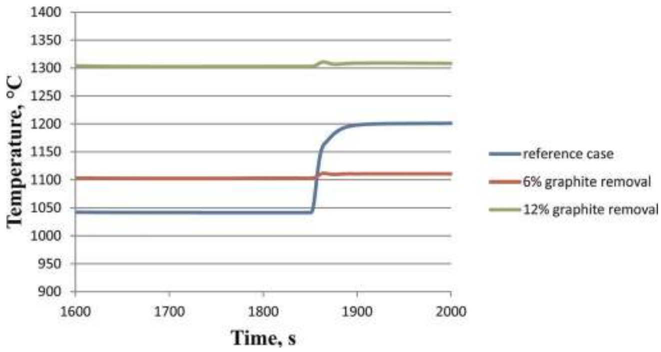

Fig. 12. Fuel temperature history of the hot channel

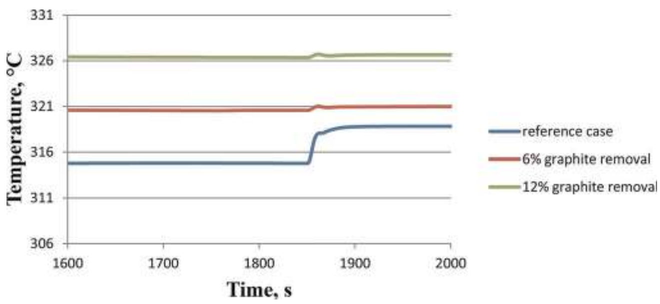

Fig. 13. Assembly cladding temperature history of the hot channel

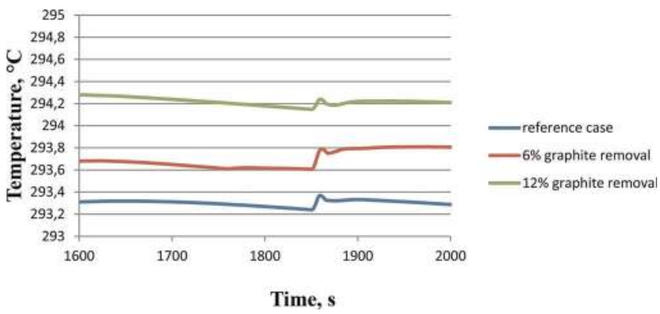

Fig. 14. Channel tube temperature in the hot channel

creases approximately by 30 MWt, and then stabilizes as the ACR rods start working.

The peripheral CPS rod withdrawal is thus also compensated by a group of ACR, and the increase of the local power in the fuel channels and the total reactor power is also partly suppressed. Moreover, the results obtained for the three different graphite removal cases are also very close to each other. As written in the previous chapter, also here no additional procedures are required to control this transient for all three graphite removal cases.

5 Conclusion

The paper analyzes two reactivity insertion transients, each taking into account three graphite removal scenarios. All analyses were performed with the coupled code system ATHLET-QUABOX/CUBBOX. The simulation results of the rod withdrawal has shown that in all three cases of graphite removal of up to 12 % of the graphite in a RBMK-1000 reactor, no additional hazards are to be expected from the design basis accidents under consideration. The study of the possibility to operate RBMK-1000 reactors in conditions of reduced graphite volume will be continued in the future bilateral work of SEC NRS and GRS. The first important validation simulations are being performed and have shown good results.

(Received on 29 February 2016)

References

1 *Langenbuch, S.; Clemente, M.; Velkov, K.*: Core calculations of the reconstructed Kursk-1 core with QUABOX/CUBBOX. Training course for SECNRS, Moscow, 2005
2 *Lerchl, G.; Austergesilo, H.*: ATHLET Mod 2.2 Cycle A. User's Manual., GRS-P-1/Vol. 1 Rev. 5 – GRS, 2009
3 *Khrennikov, N.; Danicheva, I.; Istomina, N.; Samokhin, A.; Velkov, K.; Pasichnyk, I.*: Application of 3-D coupled code QUABOX/CUBBOX-ATHLET for RBMK-1000 vapor reactivity coefficient measurements. Annals of Nuclear Energy 84 (2015) 220–224, DOI:10.1016/j.anucene.2014.12.031

List of Abbreviations

ACR	automatic control rod
CPS	control and protection system
CR	control rod
D_1	Diffusion constant in fast energy group
D_2	Diffusion constant in thermal energy group
DS	drum-separator
FC	fuel channel
FWP	feed water pump
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH
HM	heavy metal
LACR	local automatic control rod
MCC	main coolant circuit
MCP	main circulation pump
PH	pressure heard
SECNRS	Scientific Engineering Centre for Nuclear and Radiation Safety
SH	suction header
TSCV	turbine stop and control valve
TG	turbine generator
XS	cross section
β_{eff}	effective delayed neutron fraction
Σ_{a1}	Absorption cross section in fast energy group
Σ_{a2}	Absorption cross section in thermal energy group
Σ_R	Removal cross section
$\nu\Sigma_{f1}$	Fission production cross section in fast energy group
$\nu\Sigma_{f2}$	Fission production cross section in thermal energy group

The authors of this contribution

Aleksei Samokhin (corresponding author)
E-Mail: asamokhin@secnrs.ru
Scientific and Engineering Centre for Nuclear and Radiation Safety (SEC NRS)/Malaya Krasnoselskaya st. 2/8, bld. 5
107140 Moscow, Russian Federation

Matias Zilly
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH
Forschungszentrum
Boltzmannstraße 14
85748 Garching bei München, Germany

Bibliography
DOI 10.3139/124.110726
KERNTECHNIK
81 (2016) 5; page 498–503

ISSN 0932-3902

Books · Bücher

Knowledge Management and Its Implementation in Nuclear Organizations. IAEA Nuclear Energy Series No. NG-T-6.10, Published by the International Atomic Energy Agency, 2015, ISBN 978–92-0-107215-3, 52 pp., 31.00 EUR.

The objective of this publication is to share best practices and experiences based on the KMAV programme undertaken by IAEA expert teams during the period 2005–2013. These visits have involved different types of nuclear organizations. A secondary aim of this publication is to provide feedback on past KMAVs, and to provide guidance for the future development of the assessment tool(s), which will assist participating nuclear organizations with optimizing their future KMAVs.

This publication is intended for use in nuclear organizations. It is aligned with IAEA-TECDOC-1586, Planning and Execution of Knowledge Management Assist Missions for Nuclear Organizations and other supporting material in relevant IAEA TECDOCs and reports.

The history of KMAVs described in this publication includes visits to both NPPs and other nuclear organizations during 2005–2013. All KMAVs carried out during this period are considered. The details of these visits are described and analysed in Section 4 of this publication. Based on this analysis, the expansion of KMAVs is suggested in order to develop a specific knowledge management assessment tool for technical and scientific support organizations, which would:

- Support the implementation of knowledge management based on IAEA safety standards;
- Complement the current Integrated Regulatory Review Service and Operational Safety Review Team missions in the area of knowledge management;
- Support the capacity building initiatives of Member States.

It is not within the scope of this publication to make comments on the performance of individual organizations or to identify strong or weak areas of individual organizations' knowledge management. Specific inputs (including proprietary systems, processes and techniques) are not made available in this publication, and, wherever possible, feedback is made anonymous to protect the identity of participating nuclear organizations.

This publication is intended to be used by nuclear organizations that wish to use the IAEA's KMAV service. It provides an overview and observations from KMAVs for the years 2005–2013. Section 2 describes the elements, criteria and observations for KMAVs done in NPPs and R&D organizations. Section 3 lays out the elements, criteria and observations for KMAVs in nuclear educational institutions. Section 4 summarizes suggestions for organizations planning to request a KMAV and also provides several hints for the future development of the KMAV process and tools used for knowledge maturity assessment. The accompanying CD-ROM contains three knowledge management assessment tools for NPPs, R&D organizations and nuclear education institutions in the form of spreadsheets. These tools are free of cost and contain detailed descriptions on their use with the possibility to modify their content according to the needs of the nuclear organization.