Available online at www.sciencedirect.com
ScienceDirect
Nuclear Energy and Technology 2 (2016) 126–131
www.elsevier.com/locate/nucet

# Computational and experimental study of an irradiation rig with a fuel heater for the BOR-60 reactor

A.V. Varivtsev*, I.Yu. Zhemkov, A.V. Boev, O.V. Ishunina, Yu.V. Naboyshchikov, N.S. Poglyad, M.G. Sharonova

*JSC "SRC RIAR" (NIIAR), Dimitrovgrad-10, Ulyanovsk Region 433510, Russian Federation*

Available online 24 May 2016

**Abstract**

Advanced core materials and components for nuclear reactors of a new generation are tested in the BOR-60 reactor. A novel irradiation rig (IR) with a fuel heater is used for high-temperature tests of material samples. The irradiation rig features a number of advantages over ampoule-type irradiation rigs commonly used nowadays. Computational and experimental studies on an IR with a fuel heater have been conducted in the BOR-60 reactor core. The results of a dedicated methodical experiment have proved that it is possible to provide the required temperature conditions for irradiation of tested samples. MCU-RR, a precision code, was used for neutronic calculations, and thermohydraulic calculations were performed using the ANSYS CFX software system. A comparison of calculated temperature values against experimental data has shown a fit in the experimental error limits which confirms the applicability of the selected codes, models and procedures. Computational and experimental studies have also been conducted for the temperature distribution in the IR with a fuel heater following the withdrawal of the IR from the reactor and its placement in a dry cooling channel. The decay power in the IR fuel pins were calculated using the AFPA code and the temperature fields were calculated based on ANSYS CFX. It has been shown that the permissible temperature value on the fuel cladding is not exceeded in the IR withdrawn from the reactor following two-day cooling after the reactor shutdown.



*Keywords:* Reactor; Irradiation rig; Samples; Fuel pins; Heat rate; Power; Temperature; Thermocouple; Nuclear fuel; Enrichment; Coolant.

## Introduction

The BOR-60 reactor [1] is used for an extensive scope of experimental studies and diverse irradiation programs to justify the feasibility of new advanced materials and designs for some of the reactor components, the possibility for increasing the maximum burn-up of nuclear fuel, and the achievability of threshold neutron fluences and damaging doses, as well as to study the regularities involved in the radiation-induced change in the behavior of different materials [2]. Different types of irradiation rigs (IR) are used for in-pile tests.

Generally, the positions the samples of test materials are placed in are limited by the reactor core height (45 cm), though positions at the level of the end blanket regions (upper blanket—100 mm, lower blanket—150 mm) are also possible. Radially, the sample positions are limited by the assembly casing's inner flat-to-flat dimension (42 mm). Besides, a double-casing IR design is used in most cases to thermally insulate test samples from adjacent assemblies, in which case the casing's inner flat-to-flat dimension is 38 mm. The diameter of the sample-containing ampoules is normally 32–38 mm.

Normally, non-fissionable materials (steels, alloys, absorbers, and moderators) are tested at high temperatures (400 °C and 700 °C and more) in ampoule-type IRs, with samples placed in sealed ampoules, while the required sample

* Corresponding author.
*E-mail addresses:* vav3@niiar.ru (A.V. Varivtsev), ziu@niiar.ru (I.Yu. Zhemkov), shugart12@yandex.ru (A.V. Boev), iov@niiar.ru (O.V. Ishunina), n_yu_v@niiar.ru (Yu.V. Naboyshchikov), blais-sub@rambler.ru (N.S. Poglyad), shm2412@mail.ru (M.G. Sharonova).

Peer-review under responsibility of National Research Nuclear University MEPhI (Moscow Engineering Physics Institute).

Russian text published: Izvestia Visshikh Uchebnikh Zavedeniy. Yadernaya Energetika (ISSN 0204-3327), 2016, n.1, pp. 119-129.

http://dx.doi.org/10.1016/j.nucet.2016.05.010

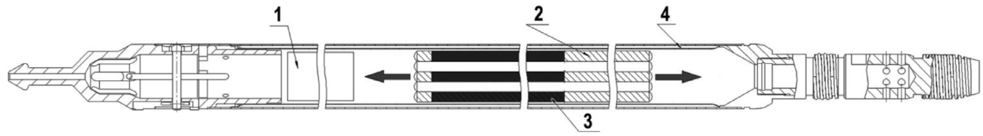


Fig. 1. An IR with a fuel heater (a longitudinal cut): 1—experimental volume for accommodation of test material samples; 2—fuel pins bundle; 3—fuel column; 4—IR wrapper.

temperature is ensured by selecting the dimension of the heat-insulating gap between the walls and the composition of the gas to fill in the gap (argon, helium, neon or mixtures thereof). Inside ampoules, samples may be contained in different environments (sodium, lead-bismuth, lead, gas and so on).

The major drawbacks of ampoule-type IRs are:

- highly non-uniform temperature distribution;
- a much smaller net volume for the test material accommodation;
- samples cannot be withdrawn from sealed ampoules for substitution or intermediate out-of-pile examination;
- the sample temperature can be only calculated (and normalized against measured values of performance parameters);
- the sample temperature is sensitive to variations in the reactor thermal power due to being weakly dependent on the coolant flow rate through the IR;
- the temperature conditions in the fabricated IR can be changed only through its relocation within the core;
- there is a risk of a gas leak not being detected by standard reactor equipment but leading to a sharp decrease in the sample temperature.

An IR with a fuel heater can be used for high-temperature tests (at 400 °C–650 °C) of the samples [3].

## Irradiation rig with a fuel heater

Fig. 1 presents a longitudinal cut of an IR with a fuel heater. Arrows show the potential directions for the fuel pins bundle displacement.

The required temperatures of samples are achieved in an IR of this type thanks to the heating of sodium on the fuel pins bundle in the IR's lower part. The heating level is ensured through the adjustment of the nuclear fuel load, the fuel enrichment, the axial arrangement of fuel pins and the sodium flow rate. Standard BOR-60 fuel can be used in the heating fuel pins. An IR with a fuel heater lacks most of the above drawbacks which are inherent in ampoule-type irradiation rigs. Thus, for example,

- test samples are contained in a reactor-grade sodium fluid heated to the given temperature, which enables uniform distribution of temperatures;
- heating fuel pins may be positioned at the level of the bottom blanket region of standard FAs and lower, this enabling test material samples to be placed along the entire core height (there are no double-wall ampoules and it is possible to eliminate the double-wall casing for an increased sample accommodation volume);
- samples are contained in a dedicated suspension withdrawable from the IR independent of the fuel bundle;
- the sample temperature is equal to the heated coolant temperature measured by thermocouples;
- the temperature of sodium and, accordingly, of samples depends on the ratio between the reactor power and the sodium flow rate which remains nearly invariable during the reactor operation at a power level close to the rated value;
- the temperature irradiation conditions can be changed both through the relocation within the reactor core and by moving the fuel heater along the IR axis;
- a potential loss of sealing in the heating fuel pins is detected by the fuel cladding integrity monitoring system. In this case, the reactor is shut down and the samples are transferred into the IR with a new fuel heater.

It should be noted that an IR with a fuel heater is more expensive to fabricate and more difficult to handle outside the reactor than an ampoule-type IR.

Therefore, an IR with a fuel heater makes it possible to achieve different sample irradiation temperatures (up to 650 °C), periodically regulate the power density in fuel pins as the fuel burns up, and, accordingly, keep the sample temperature in the given limits, and change the temperature as specified by the experiment program.

The use of this IR type in the BOR-60 reactor requires the validity of calculation programs and procedures used to calculate the in-pile IR test conditions (power density and temperature) and the IR parameters outside the reactor to be confirmed experimentally.

## Programs and procedures

Different programs and procedures described below were used for computational and experimental studies.

The BOR-60 reactor's data and measurement system (DMS) includes transducers the signals from which are processed in the computer system, stored in a special file and displayed as necessary. The DMS supports monitoring, in real time, of many reactor parameters (about 1000), as well as filing of all parameters. The DMS implements certified procedures for determination of the reactor power, the sodium flow rate and temperature, and so on. The DMS makes it possible to increase the number of measuring channels and introduce new measuring subsystems, specifically, thermocouples (TC) installed inside the IR. The DMS data are used to analyze directly measured and calculate unmeasurable reactor parameters, as well as to calculate the IR characteristics.

The BOR-60 neutronic performance automated calculation system (ACS) [4] is used to generate computational models of the reactor with regard for the actual arrangement of packages, the fuel composition, the absorber composition and the structural materials of all assemblies and control rods. The ACS allows analyzing and processing the BOR-60 neutronics, and modeling different irradiation conditions for assemblies, fuel pins and ampoules.

The IR power characteristics were calculated based on procedure in [5] enabling the contributions from the major reactor radiation components (neutron, prompt and delayed photonic) to the power density to be taken into account. The procedure was modified for the calculation of power density in the fuel. The MCU-RR [6] and AFPA [7] codes were used for the calculation.

The MCU-RR software system is designed to calculate the neutronic performance of nuclear reactors and subcritical systems by Monte Carlo method in a random 3D geometry with the energy dependence of the neutron and gamma quanta—substance interaction cross-sections accounted for in detail. The results of the calculation using the MCU-RR code have shown a good fit with data from different experimental investigations inside the BOR-60 reactor core and beyond it (in the side shield, in the rotary plug, in horizontal channels and in the biological shielding).

The AFPA code implements an analytical solution to equations describing the isotopic kinetics. The code supports calculations for decay heat, radiation characteristics (integral activity and its components, spectrum and radiation intensity), burn-up, changes in the isotopic composition of fuel assemblies in the process of in-pile irradiation, number of fissions, and liberated energy. Decay heat is calculated based on the functions of decay from the fission event on fissionable isotopes.

The thermohydraulic calculation of the IR was based on the ANSYS CFX software system [8]. The system offers broad capabilities for the analysis of hydro- and gas-dynamic processes, multiphase fluxes, radiation heat transfer and more. The following is used for analyses: finite-volume digitization of equations; solution of 3D time-dependent Navier–Stokes equations; different first- and second-order diagrams; simultaneous solution of momentum and mass equations; algebraic multiple-grid solution method for linearized equations. The ANSYS CFX system was used more than once for calculations to validate irradiation programs and methodical experiments conducted in the BOR-60 reactor.

## In-pile computational and experimental study of the IR

A methodical experiment was conducted in the BOR-60 reactor to determine the IR fuel heater power and temperature calculation error.

For the experiment, the IR was placed in an instrumented cell indexed D23 in the reactor core's row 5 (Fig. 2). The temperature in the IR was monitored continuously using two thermocouples (TC) installed immediately under the heating fuel pins. Besides, the reactor parameters (coolant flow rate and inlet temperature, reactor thermal power) were recorded using the DMS.

In neutronic calculations, the ACS was used to build a 3D computational BOR-60 model simulating the reactor state as of the start of the methodical experiment. The reactor core consisted of 119 standard and three experimental FAs, and 13 non-fuel assemblies. The BOR-60 core loading diagram is shown in Fig. 2.

The fuel column of the heating fuel pins was extending between the elevations of –48.5 and –18.5 cm relative to the median plane of the reactor core.

Fig. 3 presents the axial distribution of the linear power in the heating fuel pins with the IR placed in cell D23. The maximum value of the linear power calculated based on MCU-RR was ∼240 W/cm, and the total power of the heating fuel pins was 121 kW. The resultant values of the power characteristics were used in thermohydraulic calculations to determine the temperature of sodium over the fuel bundle.

The IR temperature calculation error depends on many factors and components. Uncertainty is introduced into the temperature estimate by the initial data determination error, including for reactor thermal power, coolant inlet temperature; sodium flow rate through the reactor and the IR; and masses and isotopic composition of fuel in fuel pins. The total absolute determination error of the sodium temperature at the fuel bundle outlet (and, accordingly, of the sample temperature) in the IR is 30 °C.

For the time of the IR irradiation in cell D23 (62 effective days), the maximum and the average calculated nuclear fuel burn-up in the heating fuel pins was 0.8% and 0.6% of heavy atoms respectively, and the power decrease was 1.1%. The calculated decrease in the sodium heating on the fuel bundle was ∼4 °C, which does not exceed the TC error.

Table 1 presents experimental and calculated coolant temperature values for the fuel bundle outlet. As can be seen from the presented data, the calculated and experimental values of temperatures at the TC positions coincide in the limits of the experimental error, which confirms the accuracy of the calculated power values.

## Computational and experimental study of the IR in a dry cooling channel

It is planned that the IR will be periodically unloaded from the reactor for intermediate studies, for the withdrawal of fluence and temperature monitors, and for the substitution of some of the samples. The IR will be withdrawn and placed back in the reactor during scheduled outages which may be long (40–45 days) or short (10–20 days).

After the withdrawal from the reactor, the IR is transferred in the air environment, washed of sodium and placed in a dry cooling channel (DCC), and is then transported to a hot chamber for being studied and handled as required.

DCCs are intended for the interim storage of irradiated standard and experimental assemblies of the BOR-60 reactor. The channels are deployed inside the concrete mass spaced at 400 mm. The DCC is designed as a vertical stainless-steel

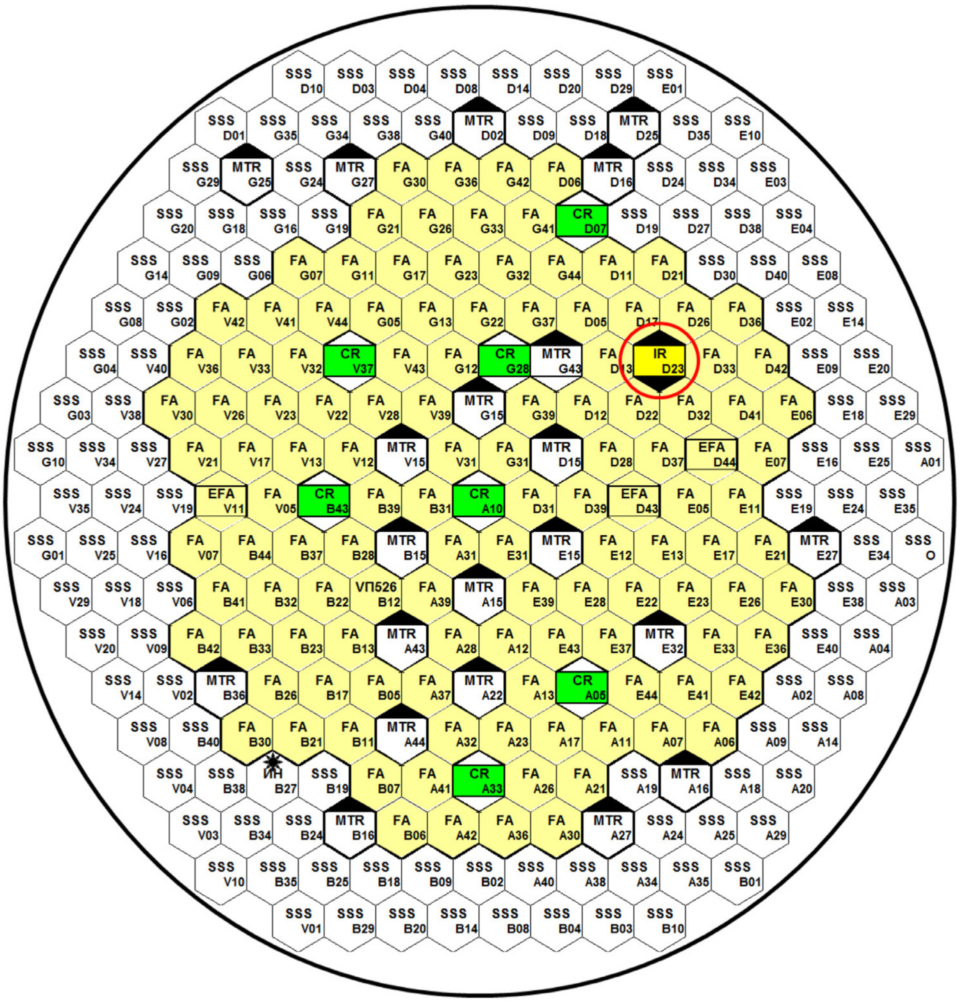


Fig. 2. BOR-60 reactor core loading diagram: FA—fuel assembly; EFA—experimental fuel assembly; CR—control rods; MTR—materials test rig; SSS—side shield steel assemblies.

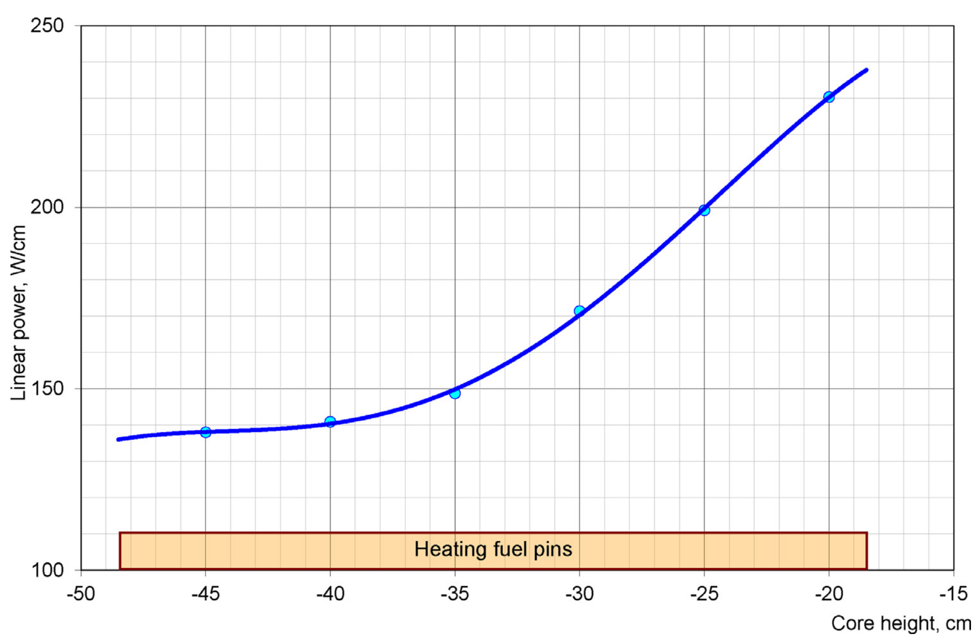


Fig. 3. Axial distribution of the IR fuel pin linear power.

Table 1
Calculated and experimental temperature values, °C.

| TC | Elevation from reactor core median plane, mm | TC | Calculation |
|---|---|---|---|
| $T_{in}$ (Na)* | –325 | 315 ± 6 | 315** |
| $T1$ | –102 | 585 ± 6 | 590 ± 30 |
| $T2$ | | 600 ± 6 | 595 ± 30 |

* IR outlet sodium temperature.
** The inlet temperature assumed to be constant for the calculation (315 °C).

tube with a diameter of 108 mm and a height of 10,500 mm. Heat is removed from the assemblies by natural air convection.

The decay heat in fuel pins and a major deterioration in the cooling conditions during the IR transfer in the air environment may cause the fuel cladding and test sample temperatures to exceed the permissible values.

When the standard technology is used, irradiated FAs are unloaded from the BOR-60 reactor not earlier than 15 days after the reactor shutdown, which ensures that the maximum fuel cladding temperature (650 °C) is not exceeded. So irradiated FAs are withdrawn from the reactor only twice a year during lengthy shutdowns.

It should be noted that the thermal power of standard FAs, depending on the irradiation position and the fuel burn-up, is 300–600 kW, and the IR power is much lower (100–130 kW). This requires a computational and experimental study to find out if it is possible to cut the time for cooling the IR with a fuel heater in the reactor after the irradiation is over, this expected to permit the IR withdrawal during any reactor outage.

The objective at this stage of the study is to determine the time for the IR with a fuel heater to be cooled after the irradiation as required for the IR to be safely unloaded from the reactor and placed in the DCC.

Table 2
Comparison of calculated and experimental IR temperature values.

| Cooling time (days) | Calculation | Experiment | |
|---|---|---|---|
| | | TC 1 | TC 2 |
| 1 | 228 ± 25 | – | – |
| 2 | 209 ± 23 | 197 ± 3 | 198 ± 3 |
| 3 | 198 ± 22 | 185 ± 3 | 185 ± 3 |
| 4 | 190 ± 21 | 177 ± 3 | 177 ± 3 |
| 5 | 184 ± 20 | 172 ± 3 | 171 ± 3 |

*Note:* Temperature determination error for a TKhA-type TC ~3 °C.

The decay power in the IR for the preset cooling times after the irradiation in the reactor is over was calculated using the AFPA code and the empirical Way–Wigner formula. The decay power calculation results are presented in Fig. 4.

It should be noted that the values obtained by Way–Wigner formula are estimated values. The AFPA results are 5%–10% higher and it is exactly them that were used as the input in the IR temperature field calculations for different cooling periods after the irradiation in the reactor.

The IR temperature calculation error depends on many factors and components, including:

– nuclear fuel mass in the heating pins, isotopic composition and axial dimensions (1%);
– calculated values of the decay power in fuel pins (10%);
– computational IR model, constants, approximations (5%).

Therefore, with regard for the independence of the error components, the total IR temperature determination error is 11%.

Table 2 presents experimental and calculated values of temperatures at the TC positions in the IR, as well as calculated values of the maximum fuel cladding temperatures.

As can be seen from the table, the calculated and experimental values fit in the calculation error limits. Therefore,

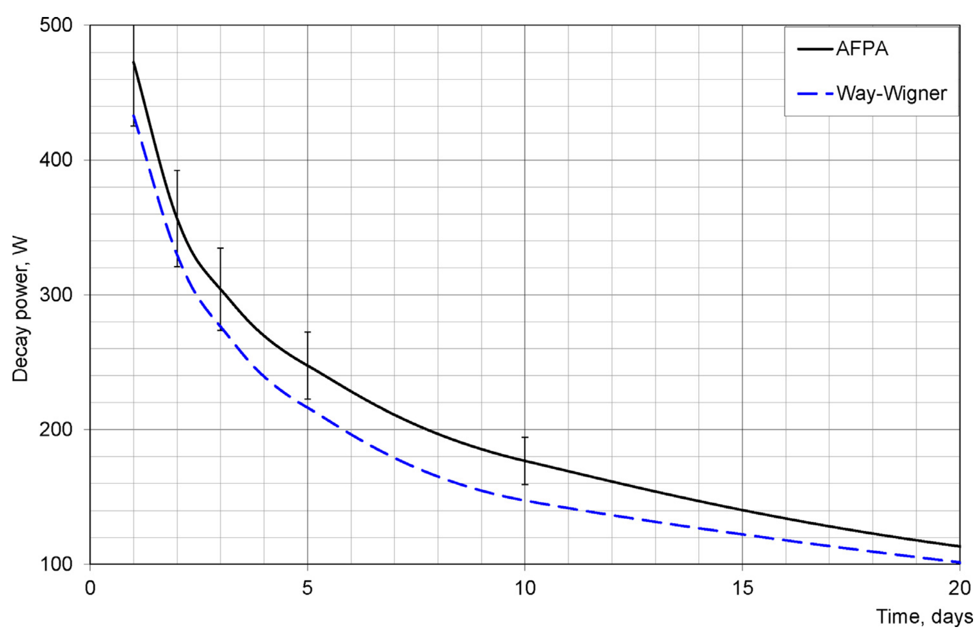


Fig. 4. Decay power in heating pins as a function of time.

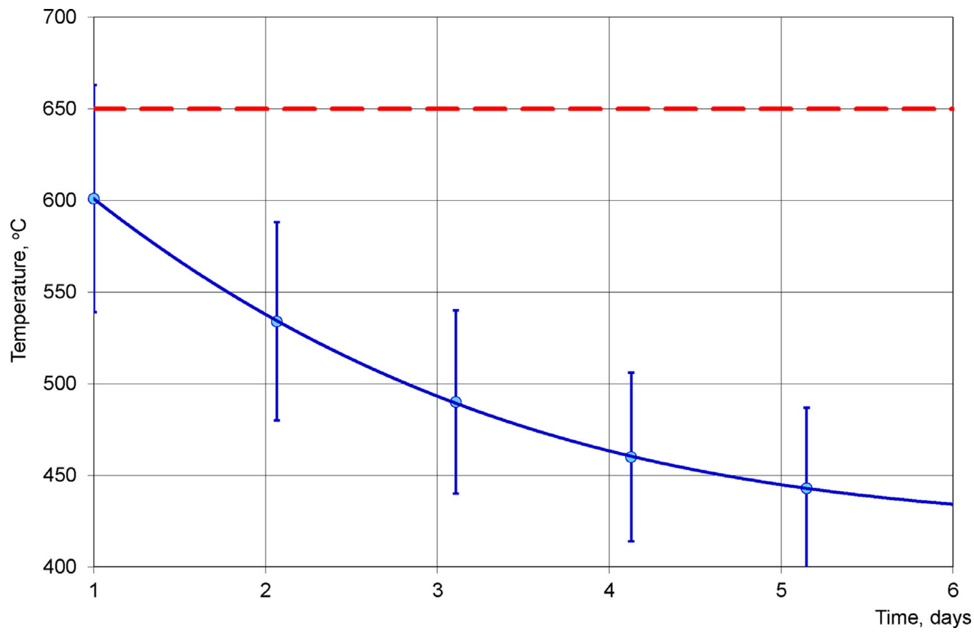


Fig. 5. Maximum cladding temperature as a function of time after shutdown.

the proposed codes (AFPA and ANSYS) and models can be used to determine the temperature conditions for the storage of IRs with a fuel heater in a DCC.

Fig. 5 presents the dependence of the maximum fuel cladding temperature on the time after the shutdown when the IR is placed in a DCC (in the air environment). The horizontal dashed line shows the permissible value of the fuel cladding temperature (650 °C).

The obtained results show that the permissible temperature value on the fuel cladding (650 °C) is not exceeded (with regard for the error) in the IR withdrawn from the reactor following two-day cooling after the reactor shutdown. Therefore, the IR with a fuel heater may be withdrawn from the reactor both during prolonged and short-time scheduled reactor outage.

## Conclusions

It has been suggested that in-pile tests of different structural materials at temperatures of 400 °C–650 °C should be conducted with the use of an IR with a fuel heater possessing a number of advantages over an ampoule-type IR.

Computational and experimental studies have been conducted to justify the conditions for irradiation of structural materials inside an IR with a fuel heater in the BOR-60 reactor. The results of neutronic and thermohydraulic calculations have shown that it is possible to achieve the required sample temperatures during irradiation in the BOR-60 reactor.

To verify the calculations, a methodical experiment was conducted in an instrumented cell of the BOR-60 reactor. The verification results have confirmed the accuracy of the calculations and the validity of the calculation procedures and programs used.

Computational and experimental studies have been conducted for the distribution of temperatures in the IR with a fuel heater where the IR is withdrawn from the reactor and placed in a dry cooling channel. The calculated and experimental values of the temperatures in the IR have shown a fit in the error limits. Therefore, the proposed calculation programs and models can be used to determine temperature conditions for the dry storage of an IR with a fuel heater and other experimental fuel assemblies.

It has been shown that it is possible to unload an IR with a fuel heater from the BOR-60 reactor two days after the reactor shutdown with no permissible fuel cladding temperature values being exceeded.

## References

[1] A.V. Varivtsev, I.Yu. Zhemkov, A.L. Izhutov, Yu.M. Krasheninnikov, Yu.V. Naboyshchikov, V.S. Neustroev, V.K. Shamardin, Nucl. Eng. Technol 47 (3) (2015) 253–259.
[2] V.S. Neustroev, S.V. Belozerov, Makarov Ye.I., Ostrovskiy Z.Ye, Fizika Metallov i Metallov. 110 (4) (2010) 394–397.
[3] Yeremin S.G., Plotnikov A.I., Zhemkov I.Yu. A device with a fuel heater for irradiation of materials in a nuclear reactor. Russian Federation Patent No. 2524683, Appl. 09.01.2013, published on 10.08.2014, Bulletin No. 22 (in Russian).
[4] I.Yu Zhemkov, Proc. Dimitrovgrad: SSC RIAR Publ. (4) (1996) 55 p. (in Russian).
[5] A.V. Varivtsev, I.Yu. Zhemkov, Phys. Atomic Nucl 77 (14) (2014) 1664–1670.
[6] Ye. Gomin, L. Mayorov, in: Proceedings of International Conference, 2, Madrid, Spain, September 27-30, 1999.
[7] V.M. Kolobashkin, P.M. Rubtsov, P.A. Ruzhansky, V.D. Sidorenko, Radiation Characteristics of Spent Nuclear Fuel, Energoatomizdat Publications, Moscow, 1983, p. 51. (in Russian).
[8] A.S. Shalumov, A.S. Vanchenko, O.A. Fadeev, D.V. Bagaev, Introduction to ANSYS. Strength and Thermal Analysis: Manual, KGTA Publications, Kovrov, 2008 (in Russian).