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SCK•CEN-BLG-1006

Qualification of the on-line power determination of fuel elements in irradiation devices in the BR2 reactor

Ludo Vermeeren, Jean Dekeyser, Philippe Gouat, Silva Kalcheva, Edgar Koonen, Vadim Kuzminov, Alfons Verwimp, Marcel Wéber

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SCK•CEN
Boeretang 200
2400 Mol
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SCK•CEN
Boeretang 200
2400 Mol
Belgium

		Date	Approval
Author	L. Vermeeren		
Verified by:	A. Verwimp		
Approved by:	J. Dekeyser		

© SCK•CEN
Belgian Nuclear Research Centre
Boeretang 200
2400 Mol
Belgium

Phone +32 14 33 21 11
Fax +32 14 31 50 21

http://www.sckcen.be

Contact:
Knowledge Centre
library@sckcen.be

Abstract

Fuel irradiation tests require an on-line monitoring of the fuel power. In the BR2 reactor, this is performed by continuously measuring the enthalpy change in the coolant of the irradiation device and complementing this information with data on power losses, heating of structure parts and spatial power profiles from mock-up test experiments and from calculations. Since a few years Monte Carlo codes (MCNP) are used, describing the BR2 core in great detail for every reactor cycle with its specific core load, yielding not only reliable relative values, but also calculated absolute local power values in agreement with data from PIE analyses. Several methods were conceived to combine the experimental and calculated data for the on-line calculation of the local linear power in the fuel elements; their internal consistency and the consistency with gamma spectroscopy data and data from radiochemical fission product analysis was checked. The data show that fuel irradiations in BR2 can be performed in a well-controlled way, with an accurate and reliable on-line follow-up of the fuel power.

Table of contents

1. Introduction

The BR2 reactor [1] has a long-standing experience in the irradiation of test fuel in various environments, both in steady state [2] and in transient conditions [3]. To monitor the irradiation conditions and to maintain them within the preset margins, the power deposited in the fuel elements (total power and peak linear power) has always been determined on-line. For this purpose the enthalpy change in the coolant of the irradiation device is measured continuously and combined with information on power losses, heating of structure parts and spatial power profiles from mock-up test experiments and from calculations.

Until recently, mainly deterministic 1-D or 2-D codes were used to obtain the necessary theoretical input [4,5]. Due to the severe geometric approximations in the models, the calculated absolute fuel powers deviated significantly from the observed values. However, it turned out that the calculated power distribution over different fuel elements (and structure parts) was quite reliable and a procedure relying only on these relative power data yielded results in reasonable agreement with post-irradiation gamma spectroscopy analyses.

Since a few years Monte Carlo codes (MCNP) are used, describing the BR2 core in great detail for every reactor cycle with its specific core load: the fuel configuration (with the appropriate burn-up values) and the irradiation devices. These calculations yield not only reliable relative values, but also the calculated absolute local power values agree reasonably with data from PIE analyses.

Several methods were conceived to combine the experimental and calculated data for the on-line calculation of the local linear power in the fuel elements; their internal consistency and the consistency with gamma spectroscopy data was checked. The data show that fuel irradiations in BR2 can be performed in a well-controlled way, with a reliable on-line follow-up of the fuel power.

After a brief survey of the irradiation devices currently in operation (section 2), this report discusses the various methods for the on-line fuel power determination (section 3), followed by a brief description of the MCNP calculations (section 4), an illustration of the method using a typical example (section 5) and a comparison of the results with PIE gamma spectroscopy data (section 6), radiochemical burn-up data (section 7) and neutron flux data (section 8).

2. Fuel irradiation devices

The BR2 reactor of the Belgian Nuclear Research Centre at Mol was put into operation in January 1963. This Materials Testing Reactor is SCK•CEN's most important nuclear facility and was operated during the past forty years in the framework of many international programmes concerning the development of structural materials and nuclear fuels for the various types of nuclear fission reactors as well as in the frame of fusion reactor research. The qualities and particular features of the BR2 reactor also allowed performing experiments aiming at assessing and demonstrating the safety of nuclear cores.

The BR2 design (fig. 1) is optimized for these utilizations and offers:

- A core with a central vertical 200 mm diameter channel, with all its other channels inclined to form a hyperboloidal arrangement around it. This geometry combines compactness leading to high fission power density with easy access at the top and bottom covers, allowing complex irradiation devices to be inserted and withdrawn.
- A large number of experimental positions, including 4 peripheral 200 mm channels for large irradiation devices. Through-loop experiments can be installed via penetrations in the bottom cover of the vessel.
- A remarkable flexibility of utilization: the reactor core configuration and operation mode are continuously being modified according to the experimental requirements.
- Irradiation conditions (temperature, pressure, environment, neutron spectrum…) representative for various power reactor types.
- High neutron fluxes, both thermal and fast (up to 10^{15} n/cm²·s).

Currently two irradiation devices are in operation for the irradiation of test fuel: i) the PWC/CCD device for testing single fuel rods in a capsule, which can be loaded in a variety of BR2 channels, and ii) the CALLISTO PWR loop occupying three BR2 reactor channels offering a range of possible experimental conditions for the irradiation of up to 9 one-meter long fuel rods per channel.

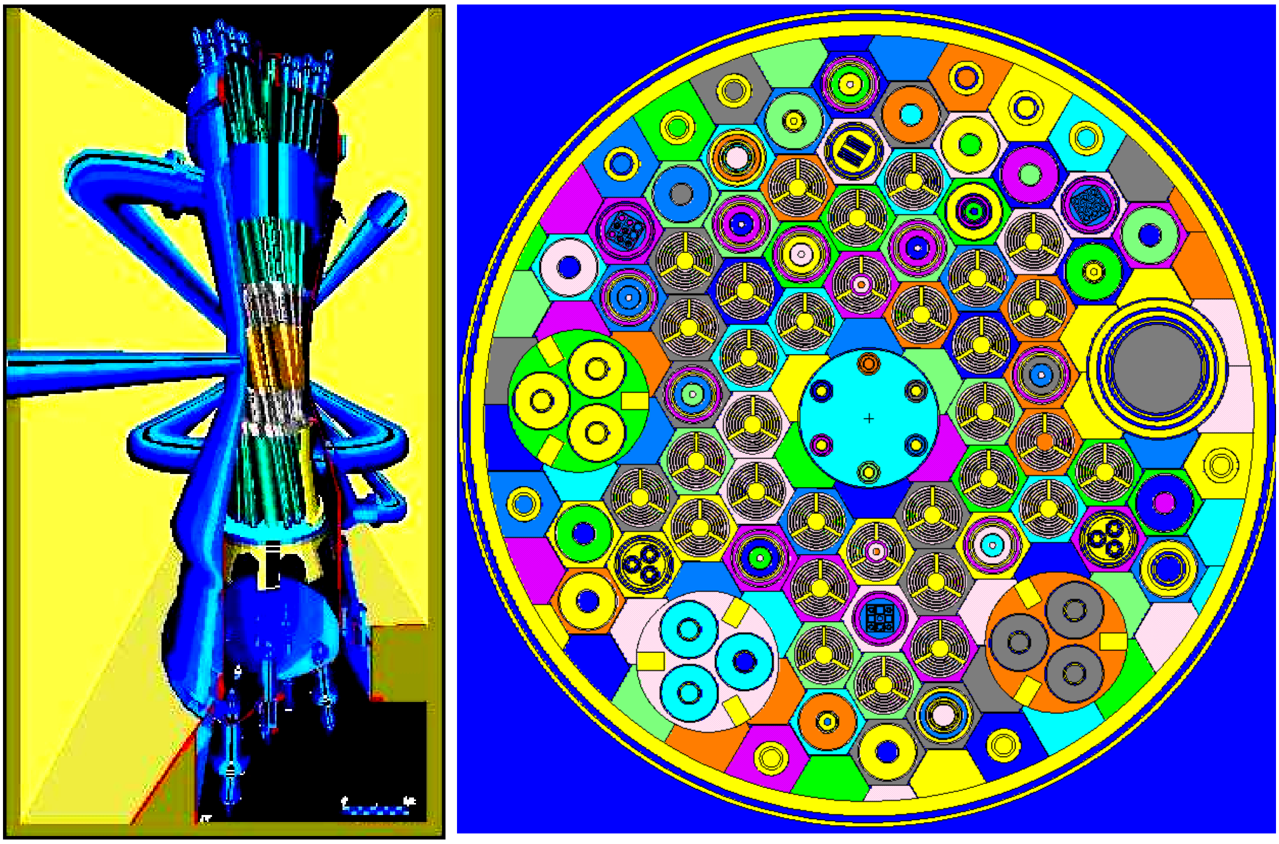


Fig. 1: Inside view of the BR2 reactor (left) and MCNP model representation of a horizontal cut 15 cm below midplane showing the hexagonal lattice with the various channels and their contents (right).

2.1. PWC/CCD

The PWC/CCD irradiation rig [6], sketched in figure 2, consists of two parts: the PWC capsule and the CCD calorimeter.

The CCD (**C**alibration and **C**ycling **D**evice) is a classic flow calorimeter allowing to monitor the thermal performance of the coolant flowing through it, using a diaphragm flow meter and thermocouples placed at inlet and outlet. The inside diameter of the stainless steel tube is about 34 mm. A 1 m high helium screen, placed on the outer surface of the CCD, serves as a thermal shield to reduce heat leaks. When using He-3 at a variable pressure it can also be used to vary the thermal neutron flux in the inner space of the CCD so as to adjust the fuel rod heating rate. This option is currently out of use due to tritium related maintenance; instead the gas gap is being filled with He-4. The fuel rod heating is adjusted by varying the BR2 reactor power during dedicated short reactor cycles. An alternative system based on a water loop with variable boric acid concentration is under development.

The PWC (**P**ressurized **W**ater **C**apsule) is an instrumented irradiation capsule for the test of single fuel rod segments with a diameter of 8-15 mm and an active length up to 1000 mm, under steady-state or transient conditions. The target fuel segment is placed into the stainless steel capsule filled with demineralized stagnant water. The water can be pressurized in the range of 0.1 to 16 MPa. The heat generated in the rod is dissipated radially through the stagnant water towards the outer surface of the pressure capsule by natural convection, with or without boiling, depending on the irradiation programme. The PWC capsule is cooled by the reactor water flow (at a typical temperature of 40-50 °C) passing through the CCD calorimeter. The normal flow rate is about 0.001 m^3/s, leading to a heat transfer coefficient of about 20 $W/(m^2\cdot K)$ on the outer surface of the capsule. Thermocouples have been installed to monitor the temperatures on the outer surface of the fuel rod at its midplane and in the stagnant water of the PWC. Water samples of the PWC water can be taken before, during and after reactor operation for monitoring the fission product activity in order to detect a possible fuel rod failure.

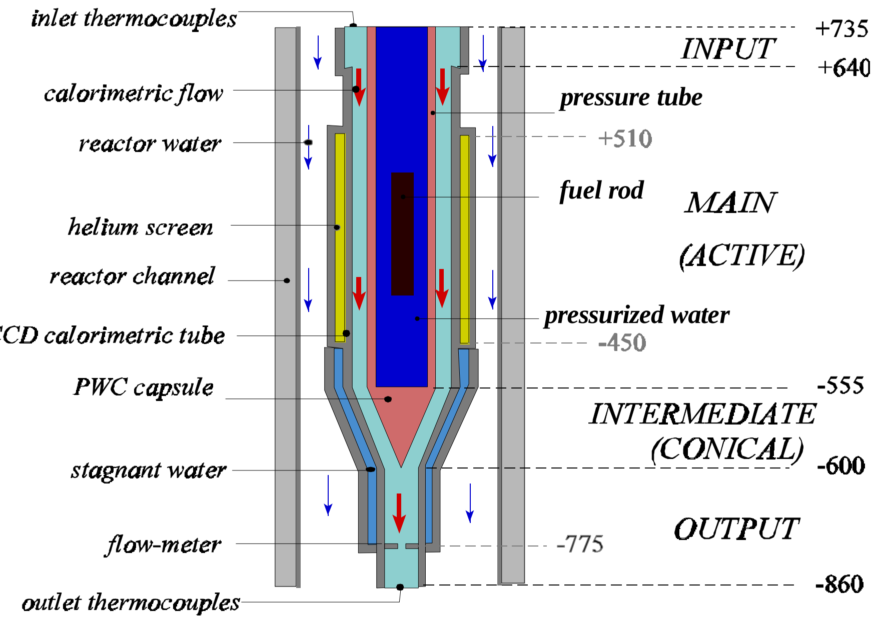


Fig. 2: Scheme of the PWC/CCD device with axial reactor coordinates in mm ('0' being the centre of the BR2 core).

2.2. CALLISTO

CALLISTO (figure 3) [7] is a PWR experimental facility for scientific in-pile studies. Three experimental rigs, called In-Pile Sections (IPS), are installed in three reactor channels to meet various irradiation conditions. They are connected to a common pressurized loop, which can deliver a wide range of pressure and temperature working regimes. These IPSs can be provided with dedicated instrumentation and be modified to perform many devoted irradiation studies, such as:

- the investigation of the behaviour of advanced fuel under representative PWR operating conditions and their qualification for save, reliable and economical use in power reactors,
- the assessment of Irradiation Assisted Stress Corrosion Cracking (IASCC) phenomena in typical light water reactor materials,
- the study of corrosion processes in fusion candidate materials,
- the characterization of the performances of high neutron dose irradiated materials for light water and fusion reactors as well as for accelerator driven systems (ADS),
- the development and qualification of new on-line in-pile detectors (like neutron and gamma flux detectors, dissolved hydrogen sensors, electrochemical potential reference electrodes…) in a high neutron flux and in a relevant thermohydraulic environment.

From 1989 to mid-1992, the CALLISTO facility was designed, constructed, tested and fully licensed. In a second stage (mid-1992 to the end of 1994), 31 fuel rods (UO_2 and MOX) were irradiated at a peak linear power ranging from 225 to 430 $W.cm^{-1}$ reaching 57 $GWd.t^{-1}$ burn-up. In 1995, we started a scientific programme studying the PWR vessel steels behaviour under irradiation. After the BR2 refurbishment (1995 – 1996) one IPS, which was moved to a BR2 channel with higher thermal and fast fluxes, became especially dedicated for corrosion and material studies. From 1997 till now, many new applications in CALLISTO emerged, e.g.: fuel testing, fusion, PWR vessel steel and ADS materials studies, corrosion studies (crack initiation, crack propagation), basic mechanical material science and modelling, detectors testing, etc.

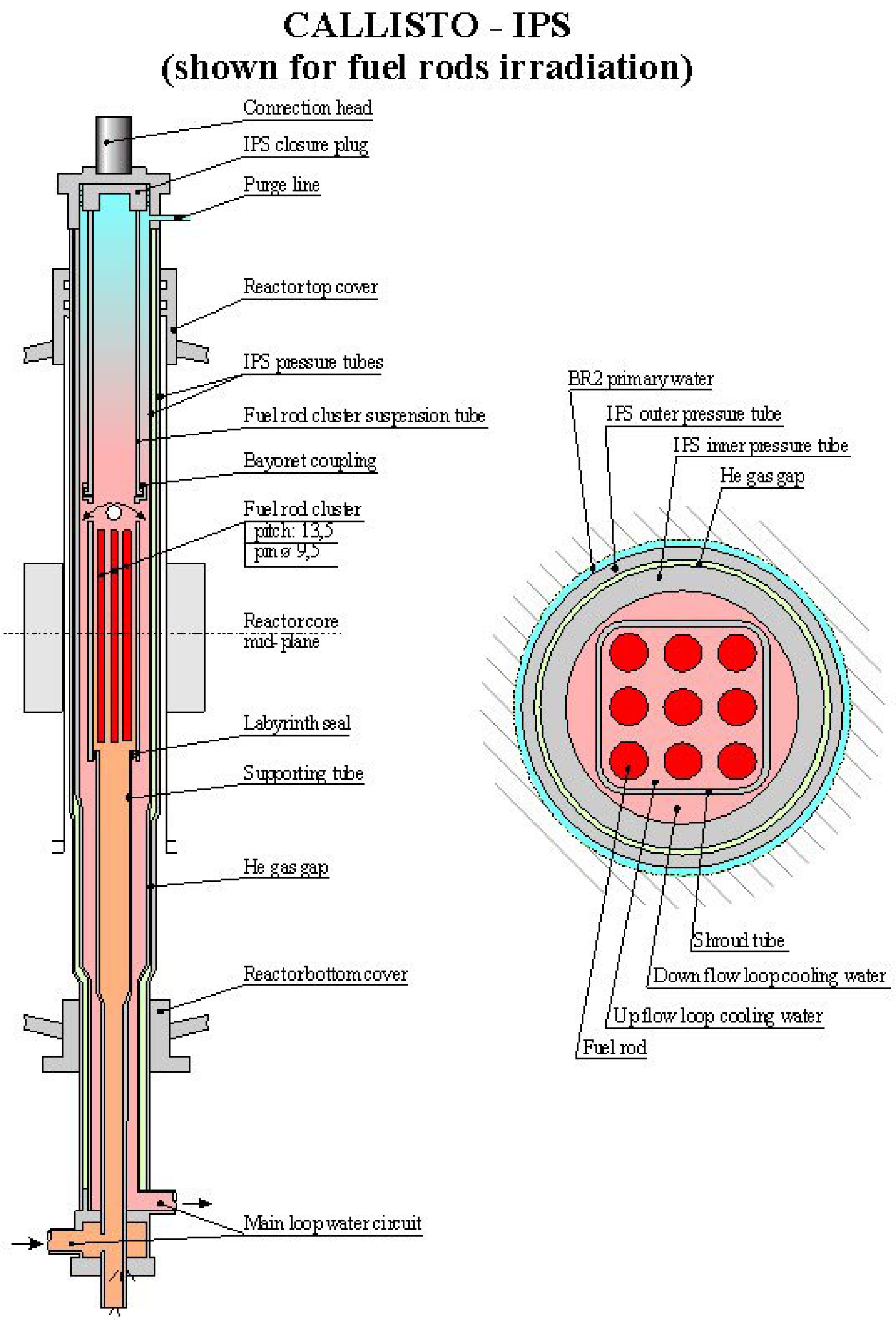


Figure 3: Typical set-up of CALLISTO IPS for fuel irradiation tests

Each IPS is provided with a basket, the capacity of which is 40.5 x 40.5 x 1000 mm^3 and it can contain fuel rods or any other targets. Alternative basket geometries with similar volume are possible. Neutron flux and gamma heating data are summarized in the following table (data at 100 % BR2 reactor power and at mid-reactor plane):

	Thermal flux (10^{14} n/cm²·s)	Epithermal flux per unit lethargy (10^{14} n/cm²·s)	Fast flux (> 100 keV) (10^{14} n/cm²·s)	Fast flux (> 1MeV) (10^{14} n/cm²·s)	Gamma heating (W/g_{Al})
IPS 1 & IPS 3	1.0 - 1.8	0.03 - 0.09	0.3 – 0.8	0.2 – 0.33	1.0 - 3.0
IPS 2	3.0 - 5.0	0.25 - 0.40	2.5 – 3.5	0.5 – 1.2	5.0 - 8.0

The axial distribution allows a 350 mm long basket section (centred around the mid-plane) to see at least 80% of the maximum flux. Depending on the fuel composition and its enrichment, the maximum linear rod power (at peak pellet) ranges from 200 to 450 W/cm.

The CALLISTO loop is operated at the following thermohydraulic conditions: an IPS inlet temperature ranging from 80 ºC to 300 ºC, a coolant pressure between 1.0 MPa and 15.7 MPa and a flow rate in the IPS in the range from 1.8 kg/s to 2.5 kg/s. The chemical composition of the CALLISTO cooling water represents that of a PWR primary circuit. Typical conditions are (allowing some variation):

- Boron (boric acid): ± 400 ppm,
- Lithium (lithium hydroxide): 1.8 ppm $\leq$ [Li] $\leq$ 2.2 ppm,
- pH: $7.00 \leq pH_{25^\circ C} \leq 7.08$ or $7.26 \leq pH_{300^\circ C} \leq 7.34$,
- Dissolved hydrogen: 25 ccSTP/kg $\leq$ $[H_2]$ $\leq$ 35 ccSTP/kg.

3. On-line power determination based on thermal balance

3.1. Thermal balance

From the equation of energy conservation (in differential form), we have:

$$dQ = G\left(dh - \frac{dP}{\rho} - dF\right) \qquad (1)$$

for the coolant flow in the irradiation device. Integration over the height (between the inlet and outlet thermocouples) leads to:

$$Q_{MEASURED} = \underbrace{G.\left[h(T_{OUTLET}, P_{OUTLET}) - h(T_{INLET}, P_{INLET})\right]}_{EnthalpyVariation} - \underbrace{G\sum_i \frac{P_{OUTLET\,i} - P_{INLET_i}}{\rho(T_{AVG_i}, P_{AVG_i})}}_{ExpansionWork} - \underbrace{2G^3 \sum_i \frac{\varsigma_i}{A_i^2 \rho(T_{AVG_i}, P_{AVG_i})^2}}_{FrictionWork}, \qquad (2)$$

where:

Q	is the power	(W)
G	is the mass flow rate	(kg/s)
h	is the specific enthalpy	(J/kg)
P	is the pressure	(Pa)
ρ	is the specific mass	(kg/m³)
F	is the specific friction work	(J/kg)
T	is the temperature	(°C)
A	is the cross section	(m²)
ζ	is the dynamic friction loss coefficient	

This equation represents the net heat flow rate to the coolant. The last two terms in the expression above are usually much smaller than the first one, so they can often be neglected. The pressure values are determined on the basis of the measured inlet pressure combined with a pressure drop along the device calculated with 1D routines.

In order to obtain the total incoming heat flow rate into the coolant during its passage through the rig, the heat losses between the coolant and the BR2 cooling water (between the inlet and outlet thermocouple positions) should be taken into account. In the case of the PWC/CCD device, these losses are very small due to the low water temperature and the presence of the He screen (and a stagnant water screen). A 2D axisymmetric thermal-hydraulic finite element calculation of all heat transfer coefficients leads to a relative loss of typically only 0.25 %.

For the CALLISTO loop, the losses can be much more important. The radial power loss is determined using an analytical model; it is essentially proportional to the temperature difference between the CALLISTO coolant and the BR2 cooling water, with a small correction term proportional to the average gamma heating rate in the device. The axial power loss at the head of the device is more difficult to model. Therefore, the total power loss is measured in the real configuration just before each irradiation cycle (at zero reactor power and as a function of the CALLISTO temperature) and corrected for the calculated radial power loss in order to obtain an empirical relation between the temperature at the highest point of the irradiation device and the axial power loss. Typical values are 10 kW for the radial power loss and another 10 kW for the axial power loss (with CALLISTO at 300 °C and the BR2 cooling water at 40 °C).
The total incoming heat flow rate into the coolant is finally determined as:

$$Q_{TOTAL}\ (\mathrm{W}) = Q_{MEASURED}\ (\mathrm{W}) + Q_{LOSS\_TOTAL}\ (\mathrm{W}) \tag{3}$$

3.2. Power deposited in structure elements

The power from the thermal balance method (Q_{TOTAL}) includes the heat generated not only in the fuel (Q_{FUEL}), but also in the structural parts of the device ($Q_{STRUCTURE}$). Both contributions must be disentangled in order to get information on the total fuel power:

$$Q_{FUEL} = Q_{TOTAL} - Q_{STRUCTURE} \tag{4}$$

During the last years significant progress was made in resolving this issue by combining detailed reactor physics calculations with experimental data from well-targeted mock-up irradiations.
Two distinct methods were implemented in INCA, the recent home-written on-line calculation software.

1. Determination of the power in the fuel from the computed relative power distribution in all parts of the rig and calibrated on the measured Q_{TOTAL}.
2. Determination of the power in the fuel from the thermal balance with the fuel loaded, after correction for the calculated heating in the structure parts (with the fuel rod loaded), the latter calculation being calibrated using the measured heating in the structure parts during a mock-up irradiation.

The two methods use different approximations and rely in a different way upon information from calculations and experiments. So, a critical evaluation of the differences between the results obtained with the two methods provides a powerful tool to assess the reliability of the methods.

First method

The first method consists in computing the relative power distribution in all parts of the device (with the fuel rod loaded) by the MCNP reactor physics code with best estimate modelling to take into account all particles transport and the delayed phenomena. Since the MCNP code outputs the results normalized per source neutron, the conversion to absolute power data is subject to additional uncertainties. Therefore, the *relative* MCNP results are considered to be more reliable.
Formally:

$$Q_{STRUCTURE} = W_{STRUCTURE} \times Q_{TOTAL} \tag{5}$$

$$Q_{ROD\_i} = W_{ROD\_i} \times Q_{TOTAL}, \tag{6}$$

in which $W_{STRUCTURE}$ and W_{ROD_i} are the calculated relative power fractions in all structure parts and in fuel rod "i", respectively. In fact, the MCNP calculations are performed for a number of BR2 control rod height values and W-values corresponding to the actual control rod height are obtained via interpolation. To obtain the average linear power of the fuel rod "i", Q_{ROD_I} is divided by the length of the fuel rod.

Second method

The second method calculates $Q_{STRUCTURE}$ according to the following expression:

$$Q_{STRUCTURE} = Q_{STRUCTURE}^{MCNP} \frac{Q_{TOTAL\_DUMMY}^{MEASURED}}{Q_{TOTAL\_DUMMY}^{MCNP}}. \tag{8}$$

This expression can be interpreted in two ways. Primo, one can consider that $Q_{STRUCTURE}$ is taken as the calculated power deposited in the structure parts ($Q_{STRUCTURE}^{MCNP}$), but that possible systematic model errors are accounted for by multiplying with a calibration factor N, obtained from a dummy irradiation:

$$N = \frac{Q_{TOTAL\_DUMMY}^{MEASURED}}{Q_{TOTAL\_DUMMY}^{MCNP}} \qquad (9)$$

An alternative interpretation of the same formalism consists in taking $Q_{STRUCTURE}$ as the total measured heating in the dummy case, but accounting for the difference between both irradiations (mainly the rod material, but also details in the BR2 configuration) by multiplying with the appropriate ratio of power data from MCNP:

$$\frac{Q_{STRUCTURE}^{MCNP}}{Q_{TOTAL\_DUMMY}^{MCNP}} \qquad (10)$$

(after proper scaling of each of the power values, according to the actual total BR2 power). Also the contribution of the stainless steel rod heating to the experimental total dummy heating is properly taken into account in this way. Of course in this method it is crucial that both MCNP calculations (for the dummy case and for the fuel irradiation) are carried out in a consistent way.
The influence of the reactor control rod height is taken into account by on-line interpolation between $Q_{STRUCTURE}^{MCNP}$ -values for various control rod heights. The semi-empirical factor N might also depend slightly on the control rod height; this dependence is determined before the fuel irradiation on the basis of the calculated and the measured data for the total dummy power at several control rod height values.
For the case of multiple fuel rod irradiation, the distribution of the fuel power (obtained via equation (4)) over the respective fuel rods should be determined on the basis of the MCNP results for the relative powers, as in the first method.

3.3. Axial shape factor

For most experimental fuel irradiation programs, not the total power in each fuel rod (Q_{ROD_I}) itself, but rather the maximum linear power ($q_{ROD\_I\_MAX}$) is the crucial parameter. To access this parameter, information on the (instantaneous) axial power profile along the fuel rods is needed:

$$q_{ROD\_i\_MAX} = \frac{Q_{ROD\_i}}{l} B_i \,, \qquad (11)$$

with l the length of the active part of the fuel rod and

$$B_i = \frac{q_{ROD\_i\_MAX}}{q_{ROD\_i\_AVG}} \qquad (12)$$

From BR2 reactor experience, some information is available on the typical axial power profiles. However, this profile depends on the reactor loading, on the positions of the control rods and on the specific properties of the irradiation device itself: the presence of the test fuel also influences the power profile. Therefore the determination of the maximum linear power in the fuel rod(s) is based on the relative power data obtained from the MCNP calculations.
The MCNP calculations yield data for the deposited power in the fuel, subdivided in axial segments of typically 1 to 2 cm length. If the resulting axial profiles for the fuel heating are sufficiently smooth, they can be fitted to determine the B-factors for every control rod position. The resulting values for B_i as a function of the control rod position are subsequently fitted with polynomial functions, which are implemented in the on-line power determination programme to determine instantaneous B_i-values and thus deduce $q_{ROD\_I\_MAX}$ data from the Q_{ROD_i} values. In a fuel configuration in which the maximum linear power is reached at a rod extremity (due to a significant end peaking effect), fitting with a smooth function (a Gaussian or a cosine function) would systematically underestimate the B-factor. In that case, the B-factor is just determined as the calculated maximum (with its statistical uncertainty) divided by the average linear power.

3.4. Uncertainty assessment

Uncertainties on various parameters will have an effect on the accuracy of the obtained values for the (maximum) linear fuel power. Considering a simplified thermal balance calculation formalism

$$Q_{MEASURED} = GC_p(T_{OUTLET} - T_{INLET}) \quad (13)$$

elucidates the most important experimental uncertainty sources:

- the differential temperature measurements: a typical uncertainty of 0.2 °C on the temperature difference between coolant inlet and outlet, i.e. a relative error of 1 to 2 % (depending on the actual differential temperature),
- the absolute coolant temperature measurement, resulting in an uncertainty of the deduced coolant heat capacity C_p of the order of 1.2 %,
- the coolant flow rate G, measured via a diaphragm flow meter with an accuracy of 2 %.

Quadratic addition of the uncertainties (considered to be independent) leads to an uncertainty on $Q_{MEASURED}$ of 2.5 to 3 %.

For the conversion to linear fuel power data, the following parameters cause additional uncertainties:

- the power loss from the irradiation device towards BR2, which is negligible in the case of PWC/CCD, but typically of the order of 20 kW (with an uncertainty of about 3 kW) for CALLISTO operating around 300°C,
- the relative MCNP results for the structure heating (in the case of the second method also influenced by the dummy irradiation data) and the average and maximum linear fuel rod powers: in view of the fair agreement of absolute MCNP power data with experimental results, the systematic error on the relative data is estimated to be at maximum 2 %. Statistical error margins on average fuel rod power data are of the order of 2 %, while those on the maximum linear power data vary between 3 and 6 % (depending on what degree of fitting can be applied).

Taking all these factors into account, one obtains a typical uncertainty of 4 to 6 % for the maximum linear power data and 3 to 5 % for the average linear power values.

4. MCNP calculations

The calculations of specific heating are being performed by the Monte Carlo code MCNP-4C [8], using the cross section data from the ENDF/B-6 file. The geometry is modelled carefully in 3D (see figure 1): the BR2 reactor geometry is completely introduced, accounting even for the inclination of the channels. The relevant irradiation channel is even modelled in more detail: the irradiation rig geometry is fully incorporated. Refs. [9-11] provide extensive descriptions of state-of-the art applications of the MCNP code for the calculation of heating distributions in the BR2 reactor, including a validation with experimental data. The calculations take into account the heating contributions due to prompt and delayed neutrons, prompt and delayed γ - rays, as well as γ - rays released in neutron capture reactions (n,γ) on various materials in the BR2 reactor. The heating in the experimental device includes contributions from particles originating from all geometrical parts of the calculational model (from the irradiation channel itself, from all other channels in the reactor core, from the beryllium reflector, from all other experimental devices present during the irradiation, from the cooling water, etc.). This also holds for the calculated fuel power. Indeed, as the total fuel heating (and the fuel temperature) is a more important parameter for the fuel behaviour rather than the fission rate, we define the total fuel rod power as the thermal power leaving the rod through the cladding, or in other words, the power deposited in the rod by fission products, gammas, betas and neutrons from the rod itself, from a neighbouring test rod or from the BR2 reactor driver fuel.

MCNP calculates the energy deposited by neutrons or by γ-rays via the formula:

$$Q_i = \frac{\rho_a}{\rho_g} \iint_{VE} H_i(E)\Phi_i(r,E)dE\frac{dV}{V}, \quad (14)$$

where (i = n, γ) is the particle type (neutron or photon), ρ_a the atom density (atoms per cm^3), ρ_g the mass density (g/cm^3), Φ_i ($cm^{-2}.MeV^{-1}$) the fluence per unit energy and $H_i(E)$ the heating response function taking into account all energy deposition processes during neutron and photon transport. The energy deposited by γ-rays can be also written as:

$$Q_\gamma = \int_{E_\gamma=0}^{100MeV} E_\gamma \Phi_\gamma(r, E_\gamma)\mu_a(E_\gamma)dE_\gamma \qquad (MeV/g) \tag{15}$$

where E_γ (MeV) is the gamma energy, Φ_γ ($cm^{-2}.MeV^{-1}$) the gamma fluence per unit energy at position r and μ_a (cm^2/g) the energy mass absorption coefficient of gamma radiation with energy E_γ at position r, taking into account all photon interactions in the medium.

MCNP-4C takes into account the transport of neutrons and photons (and, if relevant, electrons) and virtually all possible interactions of those particles; however, the code does not include processes in which delayed gammas (mainly released after β^- decay of fission products) are involved. Calculations of such contributions were performed by combining MCNP-4C with ORIGEN-S (SCALE4.3). Various modules of the SCALE4.3 system are used for the calculation of the gamma spectrum and the gamma source intensity in each fuel element (BR2 driver fuel and test fuel), with the appropriate fuel composition and burn-up. These calculated gamma spectra are consecutively introduced in a separate MCNP-4C input file as an external gamma source in the corresponding reactor channels, taking into account the radial and axial distribution of the fission power in the BR2 reactor. Finally the heating in the different parts of the experimental rig due to this gamma source is calculated.

MCNP-4C outputs heating data in terms of deposited energy per mass unit (MeV/g), normalized per Monte Carlo history, i.e. per source neutron. To convert the heating data into absolute power data, the following conversion factor is used:

$$\frac{Q_{BR2} \times \nu_f}{E_{f,BR2}} \tag{16}$$

in which Q_{BR2} is the BR2 reference power, ν_f the average number of neutrons emitted per fission (2.43 for BR2, as ^{235}U is by far the dominant fissioning isotope) and $E_{f,BR2}$ the energy deposited in the BR2 reactor per fission event. The latter is set equal to 196.4 MeV [9], the total fission energy minus the energy of the produced antineutrinos (escaping from the reactor without energy loss) plus the gamma energy released at neutron capture events in the reactor.

An important parameter in the calculation for the test fuel irradiations is the amount of energy deposited into the test fuel per fission event in this fuel – this number is needed to convert the fission rate (directly calculated by MCNP) into a value for the deposited power in the fuel. This parameter is determined using the known decomposition of the fission energy (for the relevant fissile nuclei) into its constituents: the kinetic energy of the fission products and of the emitted neutrons, betas and antineutrinos, and the energy of the prompt and delayed gammas emitted. It is assumed that the antineutrino energy is completely lost, while the kinetic energy of the fission products as well as the beta-particle energy is completely absorbed locally. Dedicated MCNP calculations are performed to determine the fraction of the gamma energy generated inside the test fuel that is also deposited inside. The standard "energy per fission" value is corrected for the lost energy assessed this way. On the other hand, gammas originating from outside the test fuel also deposit some power in the fuel, which is calculated separately (using the same calculation as for the gamma heating in the structure parts) and is included in order to obtain an effective value for the deposited energy per fission.

The statistical character of the Monte Carlo type calculations implies that the results are always subject to statistical uncertainties, which decrease only with the inverse square root of the number of source neutrons. If local information with high spatial resolution is needed (e.g. linear powers in fuel segments with dimensions of the order of 1 cm in the complete BR2 model covering about a cubic meter), the statistical errors can be significant. Typical relative statistical uncertainties are of the order of 5 % for $4 \cdot 10^7$ source neutrons (corresponding to a calculation time of the order of 2 days on present-day PCs or workstations).

5. Implementation of the method: PWC/CCD example

Recently, we have performed some power ramp tests on pre-irradiated fuel segments of about 40 cm length in the PWC/CCD device [12]. After preconditioning the fuel at the specified maximum linear power during the predefined time, the maximum linear power was increased quickly (at a rate of 100 W/cm per

minute) to the target power level which was kept during the specified stabilization period. Extensive post-irradiation experiments were carried out after unloading the test fuel from the reactor.
In the following, we describe shortly the on-line power determination procedures, illustrating the general concepts discussed in section 3.

5.1. Preparation: dummy irradiations

As a preparation for the ramp tests on the fuel segments, the PWC/CCD irradiation device was loaded with a stainless steel rod and irradiated during the BR2 cycle immediately before the ramps in order to obtain necessary data for the assessment of the heating in the structure parts of the setup. Detailed MCNP calculations provided relative power data for all structure parts, subdivided in 20.6 mm long axial sections. By normalization to the overall BR2 power, also absolute power data were obtained. The calculations were performed for several BR2 control rod heights.
The on-line power determination programme INCA provided continuously values for the measured total deposited power in the rig, based on the measurement of the temperature difference between the inlet and the outlet water and of the mass flow rate. Friction and expansion work, and radial heat losses towards the BR2 cooling water were accounted for (see equations 2 and 3).
The recorded Q_{TOTAL_DUMMY} values show a clear time evolution, which was accounted for by fitting the data (scaled with the observed BR2 power) as a linear function of the vertical position of the nearest BR2 control rod (z_{ROD_1}), although other parameters like the BR2 fuel burn-up evolution also seem to play a role. A comparison of the calculated total dummy power values for several control rod positions with the corresponding function values yielded data for the normalization factor N (see 3.2). The acquired data were described by a quadratic dependence on z_{ROD_1}, which was implemented in the INCA software:

$$N = 0.8352 - 1.3397\times10^{-5} \times z_{ROD\_1}(mm) + 5.133\times10^{-7} \times (z_{ROD\_1}(mm))^2 \qquad (17)$$

yielding N-values deviating less than 10 % from unity in the full z_{ROD_1} range of interest.
The simplification related to the fact that the variation of the measured power is attributed solely to a control rod position puts a constraint on the reliability of the second method for the determination of the heating in the PWC/CCD structure parts. Therefore, the first method is usually retained as the preferred reference method for the on-line monitoring of the fuel rod power.

5.2. Preparation: MCNP calculation results for the fuel irradiations

MCNP calculations were performed for the actual BR2 configuration of the test fuel irradiations with the control rods at levels 300, 400, 440, 480, 520, 560, 600 and 640 mm, respectively. The calculated power deposited in the structure parts (normalized to the total reactor power) was found to vary linearly with z_{ROD_1}:

$$\frac{Q^{MCNP}_{STRUCTURE}(W)}{Q^{MCNP}_{BR2}(MW)} = 207.11 + 0.02469 \times z_{ROD\_1}(mm) \qquad (18)$$

INCA combined this formula with the formula for the normalization factor N in order to deduce $Q_{STRUCTURE}$ from the measured values of z_{ROD_1} and of the BR2 power (according to the second method).
The MCNP calculations also yielded values for the total power deposited in the PWC/CCD device and hence, calculated data for the relative heating in the fuel and in the structure could be derived. The INCA programme used a cubic spline interpolation of these data (as a function of control rod position) in order to provide $Q_{STRUCTURE}$ and Q_{ROD} values according to the first method.
Finally, the MCNP calculations also form the basis of the axial shape factor determination (see 3.3). Figure 4 shows an example of the calculated linear power profile including a Gaussian fit. The fit functions were used to determine the B-factors for the eight control rod positions (as the ratio between the maximum function value and the function average over the relevant height). The resulting values for B as a function of the control rod position were fitted with a fourth order polynomial function which was implemented in the INCA programme to determine instantaneous B-values.

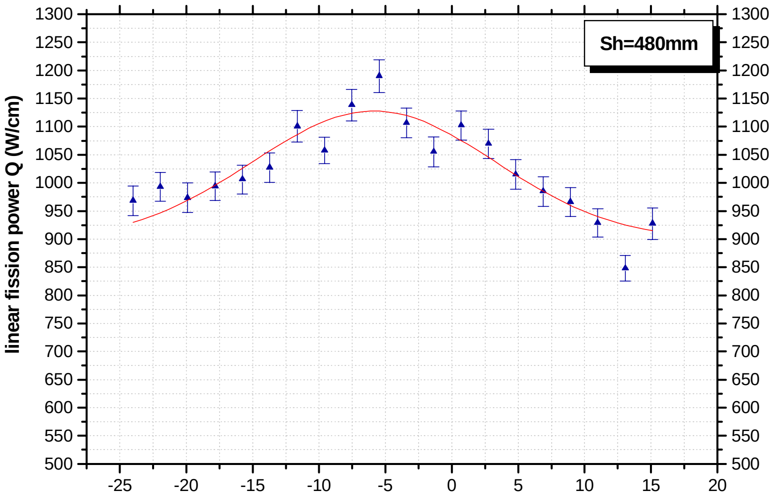


Figure 4: Example of a calculated linear power profile for a 40 cm long fuel rod in PWC/CCD with the reactor control rods at a height of 480 mm, scaled to a reactor power of 56 MW. A Gaussian fit of the data is included.

5.3. Execution of the ramps

After loading the PWC/CCD device with the test fuel segment in the appropriate BR2 channel, the reactor was started up stepwise (prescribed by the BR2 operation procedures); after some five steps, each lasting about 20 minutes, the required linear power in the test fuel rods (for the preconditioning phase) was reached. During the start-up the linear power (using the first method for the structure heating correction) was continuously monitored and the proportionality between BR2 power and test fuel linear power was verified (see fig. 5). The linear power was maintained at the appropriate level during the prescribed period. Then the BR2 power was rapidly increased by a factor determined by the ratio of the specified fuel powers before and after the power ramp. The linear power was monitored continuously and the BR2 power was controlled accordingly. After the predefined time at high power, the reactor was stopped. The PWC/CCD was discharged from the reactor and, after the required cooling time, the test fuel segment was unloaded in the hot cell. Before, during and after the irradiations, water samples were taken from the PWC to check for any indication of fuel failure.

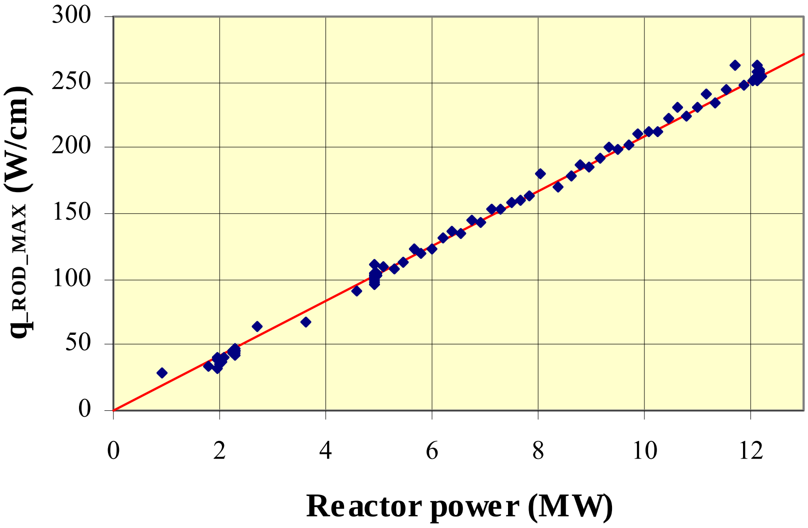


Fig. 5: Recorded maximum linear power data for a test fuel segment as a function of the reactor power during start-up (up to preconditioning conditions).

5.4. Typical on-line linear power data

The evolution of the maximum linear pin power, calculated according to both methods for the calculation of the structure heating (see section 3.2) is represented in figure 6. The maximum linear pin power values from both methods differ by 1 to 2 %, which is well within the uncertainty estimate (see 3.4).

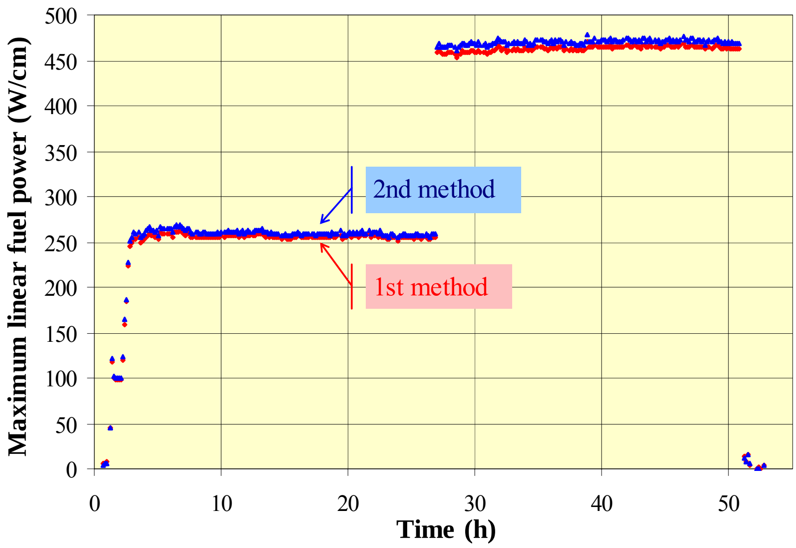


Fig. 6: Comparison of the maximum linear power data, obtained using the two different methods for the structure heating correction.

6. Validation: comparison with gamma spectrometry data

After the irradiation of each fuel segment and their removal from the reactor, gamma spectroscopy measurements can be performed. The measured activity of certain isotopes of suitable half-life (e.g. ^{140}Ba, ^{140}La), taking into account the relevant part of the irradiation history, provides data on the time-averaged fission rate during a certain period or on the instantaneous fission rate for a certain reference reactor power. Using the appropriate effective energy per fission (see section 4) these data can be converted to linear power data, which can be compared with the on-line linear power data.

The table below presents such a comparison for four fuel rod irradiations in the PWC/CCD device [13]. The third column summarizes the obtained (average) fission rates, normalized to the BR2 powers quoted in the second column. These fission rates were converted into total fuel power values and subsequently into linear power data. The average linear fuel power data from the on-line power determination are collected in the next column and the percentage difference values are quoted in the last column. The data are in very good agreement, especially in view of the assessed uncertainties of both methods: 4.2 % for the on-line power determination and 2.7 % for the gamma spectroscopy method. Moreover, the fact that the gamma spectroscopy data are slightly higher is qualitatively consistent with the presumed biasing of the gamma spectroscopy results by + 4 % (on the basis of previous intercomparison exercises [13] with various analysis methods).

Fuel rod number	Reference BR2 power (MW)	Fission rate (s^{-1})	Total fuel power (W)	Average linear fuel power after the transient (W/cm)		
				Gamma spectr.	On-line power det.	Difference (%)
1	14.0	$3.924 \cdot 10^{14}$	12510	301.3	297	+ 1.4 %
2	20.5	$5.797 \cdot 10^{14}$	18482	446.3	425	+ 5.0 %
3	20.7	$5.537 \cdot 10^{14}$	17653	423.5	426	- 0.6 %
4	16.6	$5.068 \cdot 10^{14}$	16158	388.7	369	+ 5.4 %

A comparison of the maximum linear fuel power is more difficult, since the gamma spectroscopy only yields time-integrated information and the axial power profile is expected to vary during irradiation, both with respect to width and to position of the maximum. Due to the resulting smoothed axial profile of the detected fission products, axial shape factors deduced from gamma spectroscopy will systematically be lower than the real instantaneous shape factors during the irradiation. Nevertheless, an attempt was made to reconstruct the integrated axial profiles by folding the calculated (control rod position dependent) profiles with the observed control rod position evolution. The resulting axial shapes are in good agreement with the axial fission power profiles obtained by gamma spectroscopy: an average axial shape factor for the four tabulated fuel rods of 1.072 was calculated, compared to an experimental value of 1.080 from gamma spectroscopy.

7. Validation: comparison with radiochemical burn-up analysis

The most accurate determination of the integrated fission power produced by nuclear fuels relies on destructive radiochemical procedures [14]. After dissolution of the fuel, the concentrations of selected fission products like ^{137}Cs, ^{144}Ce and several neutron-rich stable neodymium isotopes are measured, as well as the uranium, plutonium and transplutonium isotopic compositions. The extensive data are treated according to a procedure with several consistency cross check possibilities, increasing as well the reliability as the accuracy of the results. The uncertainty margin on the weighted average fission yield data ranges from 2.5 to 4 % (1s).

No radiochemical burn-up measurements have yet been performed on BR2-irradiated test fuel since the use of the new MCNP-based thermal balance method. However, radiochemical burn-up determination campaigns in the past have revealed a very good agreement with gamma spectrometry data [14]. A comparison with on-line data from the thermal balance method (which was at that time based on input data from analytical model calculations) for four test fuel rods [14] shows a typical average overestimation of the thermal balance data by 2.4 %, with a standard deviation of 7.6 %. In view of the good agreement of the recent thermal balance data with gamma spectrometry data (see section 6) it can be expected that also the deviations between the new thermal balance data and radiochemical burn-up analysis data will be smaller than for the irradiations in the past.

8. Complementary data from neutron flux detection

In order to improve the monitoring of the irradiation conditions, the irradiation device can be equipped with self-powered neutron detectors and/or activation dosimeters.

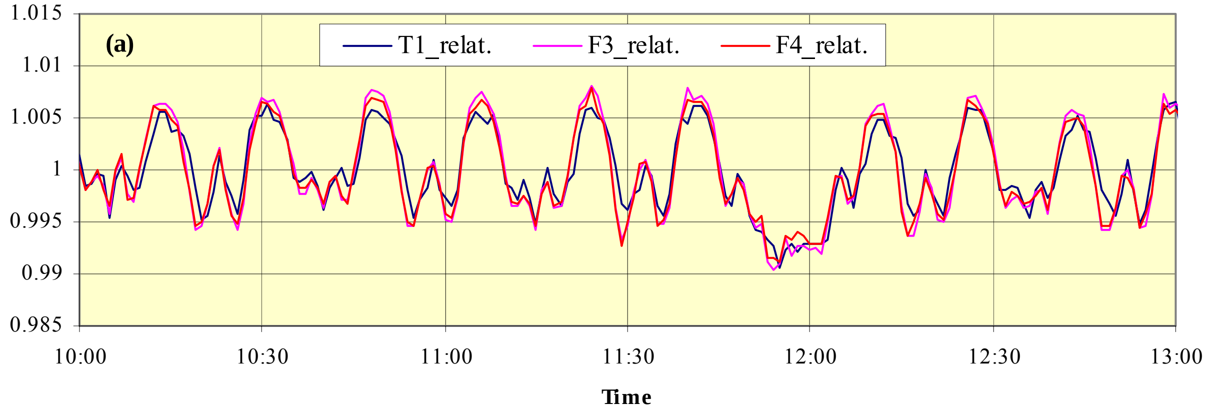
Self-powered neutron detectors (SPNDs) are small cylindrical sensors (length between 5 and 20 cm, diameter of 1.5 to 4 mm) that can be operated in in-core conditions. They output a current which is essentially a measure for the thermal neutron flux. Depending on the type of SPND the response is either immediate (with a response time below 10 ms) or delayed (response time of the order of a few minutes). In the latter case the signal can be unfolded numerically using the known response function so as to obtain an effective response time of less than a second (at sufficiently high data acquisition rate).

The MCNP calculations that yield information on the deposited powers can simultaneously produce thermal neutron flux data at the locations of the SPNDs. Consequently the experimental SPND data can be used to verify the MCNP calculations: the calculated absolute flux/power in the irradiation device (normalized to reactor power) can be checked, as well as the dependence of (relative) local flux/power data on external parameters (changing control rod positions, modification of the reactor charge, etc.). A typical example can be taken from a recent instrumented fuel irradiation in the CALLISTO device: data were collected on-line from three SPNDs (two with a rhodium emitter and one with a vanadium emitter) which were positioned in-between the fuel rods. We developed an MCNP-based model to convert the SPND currents into data on the neutron capture rates in the emitters (or, equivalently, conventional thermal neutron flux data), taking into account current contributions due to impinging gamma rays and due to high-energy beta rays directly from fission products in the test fuel. The resulting capture rate data were compared with MCNP data: the ratio between MCNP data and experimental results was found to be 0.95, 0.89 and 1.12 for the three SPNDs, respectively. In view of an estimated 10% uncertainty in the conversion from SPND current to neutron capture rate and a similar total uncertainty on the MCNP capture rate data, the SPND data confirmed the validity of the MCNP calculations within these limits.

One of the advantages of SPNDs lies in the fact that the relative accuracy of their data is considerably higher than that of the power measurements (noise level below 0.1 % in typical conditions), so changing irradiation conditions can be followed more closely. This is illustrated in figure 7, representing partial data from the instrumented fuel irradiation mentioned above. The periodical movement of a strongly absorbing rig in a nearby BR2 channel (silicon doping device) induces small changes in neutron flux and fuel power in CALLISTO. These changes were recorded via the central temperature of the instrumented fuel rods, via the signals from two rhodium SPNDs (corrected for their delayed response) and via the measured thermal balance. Figure 7(a) shows a very good correlation between the central fuel temperature data and the thermal neutron flux data from the SPNDs (correlation coefficients of 0.95). On the other hand, the measured total rig power (thermal balance method) also follows somehow the same pattern, but is subject to much more noise (figure 7(b)), leading to a correlation coefficient with the central fuel temperature data of only 0.39.

When performing fast power transient experiments, SPND data will also provide data on the shape of the power ramp with a higher time resolution, since their response is faster than that of the measured thermal balance which involves the equilibration of temperature profiles.

Activation dosimetry wires on the other hand give information on time-integrated neutron fluxes which can be coupled to time-integrated power data using the MCNP calculation results. They provide an independent verification possibility of the integrated absolute power [15].



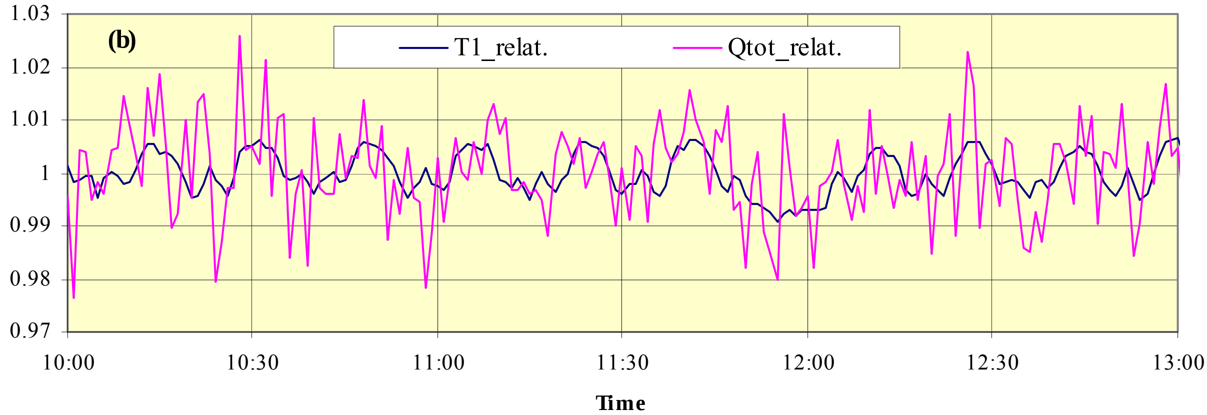


Figure 7: Normalized data from instrumented fuel irradiation in CALLISTO: central fuel temperature data (T1), rhodium SPND signals corrected for delayed response (F3, F4) and total rig power data from the thermal balance (Qtot). The temperature and SPND data follow closely small changes in irradiation conditions, while the noise on the thermal balance data (also only of the order of 1%!) masks the detailed time evolution pattern.

9. Conclusion

BR2 is equipped with irradiation devices for irradiating test fuel in various environments, for long term burn-up accumulation tests as well as for ramp tests on fresh or pre-irradiated fuel elements. The desired irradiation regime is often specified by the client in terms of maximum linear power. Therefore the maximum linear fuel rod power is continuously monitored for properly piloting the experiments. We obtain the linear fuel rod power data via a measurement of the enthalpy change in the coolant of the irradiation device, combined with validated 3D Monte Carlo core calculations (MCNP). On-line power data from recent experiments have been confirmed by post-irradiation gamma spectroscopy experiments and radiochemical burn-up analysis, proving the validity of the on-line procedure.

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