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USE OF DETECTORS FOR REACTOR IN CORE MONITORING

S. I. Azarov, A.V. Taranovski
Institute of Nuclear Research NASU, Kyiv, Ukraine
Corresponding author: azarov@kinr.kiev.ua

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Abstract
In this analytical survey of domestic and foreign scientific and technical literature publications, research problems of different types of in-core detectors and of In-Reactor Control Systems (IRCS) are considered. The problems of IRCS development are discussed and a novel non-conventional concept of reactor-control is suggested.

Keywords
reactor control system, in-core detectors, calorimetric detector, energy release control

1.Introduction

Effectiveness and reliability of nuclear reactors operation depend to much extent on their thermal stress therefore the problems of justifiable and reliable monitoring of power generation are of utmost significance. We refer to instrumentation providing in core power distribution monitoring of In-Reactor Control System (IRCS). Namely we distinguish in IRCS detectors, communication equipment, and distributed information-management with software.

The aim of this paper is to summarize and classify the world's Nuclear Power Plants (NPP) data on methods and means of in core reactor control and a brief outline of new type of in core calorimeter detectors is suggested,

2. Classification of methods and means of in core energy release control

The following classification of methods and means of in core energy release control is suggested (Fig.1):

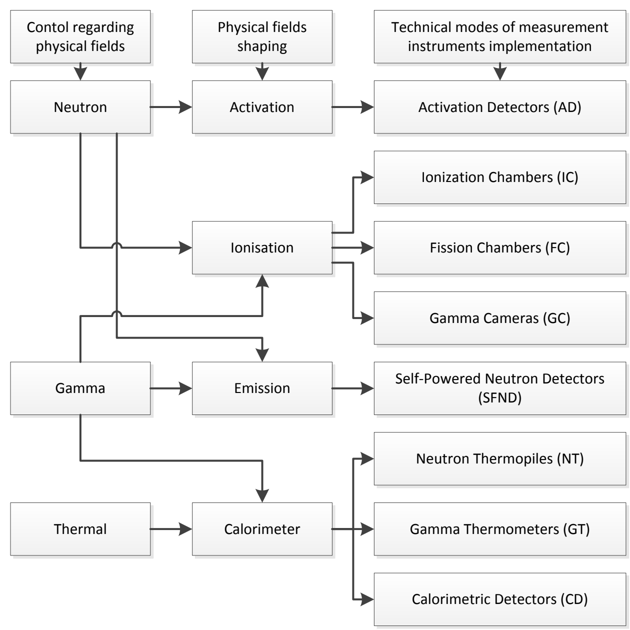

Figure 1. In-core detectors classification

- control regarding physical fields (thermal, neutron and gamma);
- by physical fields shaping (activation, ionization, emission charge, calorimeter);
- by technical modes of measurement instruments implementation: Ionization Chambers (IC), Self-Powered Neutron Detectors (SFND), Activation Detectors (AD), Fission Chambers (FC), Neutron Thermopiles (NT), Gamma Cameras (GC), Gamma Thermometers (GT), Calorimetric Detectors (CD);

- by mode of time and spatial in core energy release characteristics measurement: distribution, point distribution, quasidistribution, occasional and permanent action, stationary mounted and relocated, detector units and combined trains, e.g.

We assume under activation detector a transducer capable of ionization radiation detection in the material sensitive element with induced radioactivity occurrence at radiation exposure. Ionization detector is a transducer which function principle is based on ionizing effect in the substance of sensitive element at radiation exposure. Emission detector is a unit where potential difference between two insulated electrodes serves charged particles transfer produced at radiation exposure.

Calorimetric detector highlights heat energy disposal in material sensitive element following energy transfer. We assume under In core Detector Unit (IDU) primary transducer of measurement means distributed in reactor core subjecting radiation energy conversion into energy kind suitable signal online generation and transfer to secondary instrumentation. Methods and means of in core energy release distribution are viewed regarding the above mentioned classification.

3. Evolution survey development of methods and means for energy release control

For flexibility of retrospective analysis methods and means of energy release control development in the period 1960-1992 years are divided into three items: early on-going efforts, extensive development and stabilization period in NPS engineering 0.

Early on going efforts in NPS's development encompass (1960- 1968) years, featured pilot-industrial reactor designs, power capacity (100-300) MWt where design criteria, scientific and technical approaches validation had been realized to its extent. By design, operation parameter and functional performance nuclear reactors differed much. They had insignificant in core geometry dimension low thermal and energy stress. It accounted for lack of available technology equipment production and low operational experience in reactor operation. Therefore in the given period for each NPP an individual design of IRCS had been developing and multiplicity of methods and means of in core reactor control were distinguished. In the period of question for radial reactor monitoring a range of thermal, axial power distribution in the core, neutron and gamma activation, ionization, charge and calorimetric methods credited the existence. As activation detectors there utilized distribution periodical mode AD with wire or ball sensing elements (Mn^{55}, Cu^{63} and W^{186}) 0.

In PWR, BWR and WWER reactor types availed stationary mount measurement racks comprising FC, SPGD and NT 0. In PHWR availed GT 0 sensor units and in PWR SPGD 0. Information from in core detector units was submitted to measurement and management complex usually comprising second generation computer. Measurement systems assured signals scrap, normalization and generalization, provided computation of energy release distribution determination in 1oss-of-coolant, etc.

The stage of intensive NPP engineering development 1968-1976 years is characterized by commercial reactors development middle power capacity (400-900) MW twitch modified technical and economical qualities. Nuclear Power Plant operation experience in this stage not dubiously demonstrated that performance and physic-technical reactor make-up was not realized in full, and way of physical profiling, technological and mode modification in the same in core volume a significantly large energy stress may be produced.

On the other side nuclear reactor performance parameters build up comparing in core geometry dimension increase resulted in thermo technical and efficiency increase in NPP consequently raised economic qualities. In the given period in core volume increased on the average by two folds, energy stress by 1,5 folds and lose-of-coolant factor decreased by 1,3 folds.

Hereby it much contributed to energy release spatial instability distributions, sharp temperature and thermo stress in core deviations, following availability credit failure in core reactor instrumentation. In the period in question for reliable and available energy emission control neutron method was widely used. So far as already tested in core detector units did not meet the growing requirements to middle power capacity reactor operation, there appeared an urgent need in new typed of in core detector unite development. By the value of construction design, production technology and reliability SPND and FC stand the requirements of in core performance to high extent 0. Thus in the USSR in WWER-440 reactor SPND with (Rh, V) emitters performed for neutron spectral the characteristics determination, following signals transmittance to "Gindukush" measurement system 0.

In reactor RBMK in IRCS for axial energy emission control point SPND (Ag) were subjected (7 in number) distributed in measurement channel and 117 radial SPND. Detector signals were transmitted to "Skala" system with VNIEM computer type 0. In BWR reactors relocated and stationary distributed FC and in PWR SPND were utilized 0. In this stage sensors were performed in conjunction with third generation computers M-60, M-6000 M-7000 (USSR), Produc-250 (WE, USA), IBM-130 (IBM, USA), Ge/PAC-4020 (GE, USA), C 90-40, TAC (CAE, EEC, France), AEG 60-50 (Simens, FRG),assuring to produce nuclear-physical thermodynamic computation, of technical and economical standards for NPPs, optimization of static and transient nodes, etc.

Stabilization period (1976-1992) years was distinguished commissioning in operation reactors with sufficient single power capacity (1000-1300) MWt where in order to reduce capital investments a great part of work has been done considering enhancement of technical-economical evaluations of reactor operation, following tendency of in core instrumentation dimensions increase, performance parameters and energy stress build up.
There also appeared an urgent need in qualitatively new type of IDUs so far as traditional ones outdated themselves. Due to varieties in production technology and existing allowances at units assembling IDUs differed much in primary sensitivity and needed pre-operational calibration. On the other hand in operation performance SPND and FC exhibited change in sensitivity at radiation exposure (following Rh^{103} $Ag^{109}$and U^{235} burn up) and needed permanent detector sensitivity control change in time 0.

That's why in Ignalina NPP (RBMK-1500) as radial and axial detectors availed SPND with GC emitters which substantially changed IRCS approach development 0. In foreign NPP a wide application in IRCS found SPND with low burn up emitters fabricated of Pt, Er, Co, Yo, Ta, and Cd 0. Except neutron method a considerable amount of application was given to gamma method, where primary transducer GT was used 0.

A wide application of gamma method for energy emission control nowadays is accounted for energy in core stress in NPPs operation within high density of working media where gamma radiation serve to reliable accuracy control of energy emission distributions, being clearly justifiable among other control modes 0. At output signals treatment transmittance from IDUs decentralized mini computers are widely used, assigning them to imposed functional specifications permitting expert systems formation.
IRCS stand high requirements to performance and reliability. They should maintain energy release control by some anticipated channels provide automatically self-diagnostics of element system performance, function in normal operation and incident condition. Modification and updating in IRCS is achieved by way of modern microprocessor and programmable logics equipment disposal serving anticipated operational occurrences diagnostics within elements of control expert systems and human-machine interface improvement. In the Institute for Nuclear Research of the Ukrainian Academy of Sciences, a considerable work had been done in the field of IRCS development based on neutron and gamma method of energy release control where for in core detector instruments Calorimetric Detector (CD) are used. (see Fig.2)

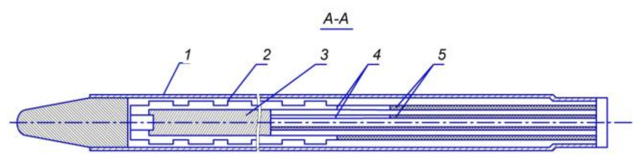

Figure 2. Calorimetric Detector: 1 - heater element; 2 - thermal sensitivity element; 3 - gamma sensitivity body; 4 - hot junction; 5-mineral insulated cable.

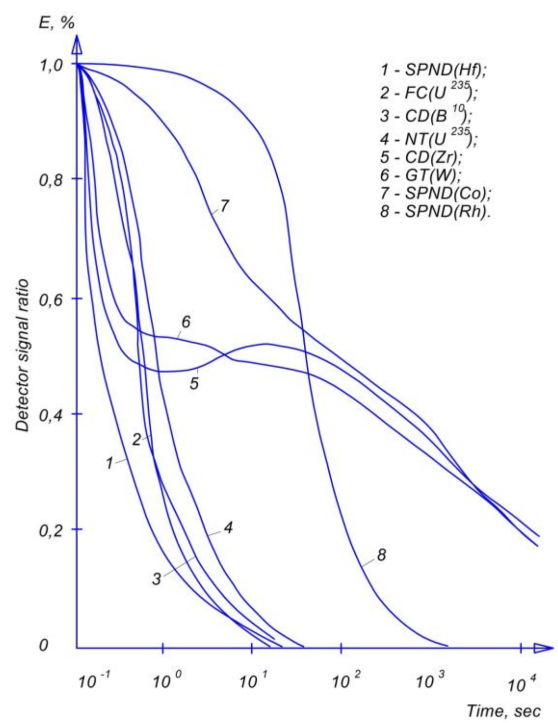

Figure 3. In core unit transients at control rod insertion.

Table 1. Comparative characteristics for neutron detectors 0

Parameter / IDU type	Size, mm x mm	Material of sensitive element	Measurement range, $m^{-2} \cdot sec^{-1}$	Sensitivity, $A \cdot m^2 \cdot sec$ ($V \cdot m^2 \cdot sec$)	Maximal temperature, K	Absolute error, %	Integrated flux, m^{-2}
IC	25 x 300	BF_3	10^{14}-10^{18}	10^{-24}	523	4	10^{24}
FC	30 x 76	U^{235}	10^{12}-10^{18}	10^{-22}	673	3	$0.2 \cdot 10^{26}$
SFND	3.0 x 200	Ag	10^{16}-10^{18}	$1.9 \cdot 10^{-23}$	600	3	$0.5 \cdot 10^{26}$
	1.8 x 200	Rh	10^{16}-10^{18}	$2.1 \cdot 10^{-23}$	800	2	$0.8 \cdot 10^{26}$
	1.5 x 10	Hf	10^{16}-10^{18}	10^{-19}	870	3	$0.7 \cdot 10^{26}$
NT	2.0 x 60	U^{235}	10^{14}-10^{16}	10^{-7}	900	5	$0.4 \cdot 1O^{25}$
CD(n)	4.0 x 60	U^{235}	10^{11}-10^{18}	10^{-8}	1000	4	$0.6 \cdot 10^{26}$
	4.0 x 60	B^{10}	10^{10}-10^{16}	10^{-7}	600	4	$0.2 \cdot 10^{25}$

Table 2. Technical characteristics for gamma detectors 0

Parameter / IDU type	Size, mm x mm	Material of sensitive element	Measurement range, Grey $\cdot$ sec^{-1}	Sensitivity, $A \cdot sec$ ($V \cdot sec$)	Maximal temperature, K	Absolute error, %	Maximal exposure, Grey
GC	6.0x200	He	0.3-1.0	10^{-14}	600	3	10^9
SPGD	3.0x600	Pt	3.0-100	$5 \cdot 10^{-12}$	770	5	$8 \cdot 10^9$
	3.0x600	Co	1.0 - 100	10^{-12}	700	4	$0.7 \cdot 10^{11}$
GT	7.0 x 100	Fe	1.1 - 100	10^{-4}	370	6	$0.3 \cdot 10^{12}$
CD(γ)	4.0 x 60	Bi	0.1 - 200	10^{-5}	500	3	10^{10}
	3.5 x 40	Zr	1.5 - 500	10^{-3}	900	5	10^{11}
	4,5 x 100	Fe	3.0 - 300	10^{-3}	600	5	$0.5 \cdot 10^{12}$
	4,0 x 50	W	10 - 400	10^{-2}	1000	7	$0.8 \cdot 10^{12}$

Calorimetric detectors are aimed at neutron or gamma radiation registration in-core, experiencing high temperature, vibration and severe ionizing radiation exposure. Allowing spectral selective sensitivity to varieties of ionizing radiation exposure (by various sets of gamma and neutron sensitive material disposal: B^{10}, U^{235}, Th^{232}, U^{238}, Np^{237}, Pu^{235}, Bi, Pb, Zr, Fe etc.), the mentioned type of reactor control instruments is considered in systems of in core energy release monitoring 0. In Fig. 3 transients for various detector types at emergency control rod insertion are shown, where SE is detector signal in % 0.

Calorimetric detectors provide its existence by high reliability and relatively simple design, performance at wire isolations resistance up to 10^3 Ohm($T \leq 1000$ K), self-calibration assurance in operation. Comparative characteristics for neutron and gamma IDUs are introduced in Tables 1 and 2. All above mentioned items allow considering this type of in core instrumentation as the most promising for high temperature reactor disposal. Calorimetric gamma and neutron detectors staged full scale performance testing on WWER and RBMK reactor types.

4. Gamma method for energy release control

The problem of power density by the absorbed dose rate determination was investigated in the cylinder like material of assigned dimension disposed in-core (see Fig. 4). The following assumptions were followed:

- homogeneous in core environment;
- induced γ-radiation neglecting;
- γ-radiation interaction with cylinder body is maintained in electron equilibrium condition.

Then the value of cylinder body overheat in quasi stationary pile mode will be determined by formula [28]:

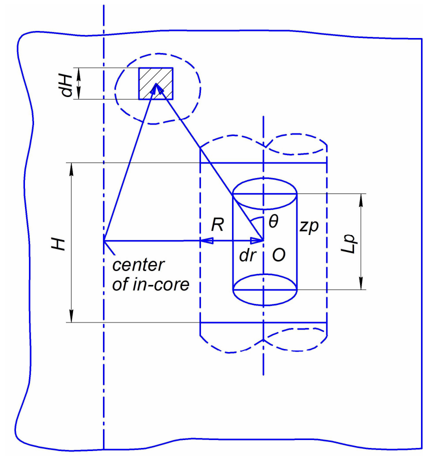

Figure 4. Schematic diagram of calculation model

$$q_\gamma\left(r_p,E_\gamma,t\right)=Cp\int_t\int_{E_\gamma}\sigma_p\left(E_\gamma\right)\varphi_\gamma\left(r_p,E_\gamma,t\right)E_\gamma dE_\gamma dt \qquad (1)$$

$$\sigma_p\left(E_\gamma\right)=\mu_p\left(E_\gamma\right)\rho_p^{-1}\cdot f\left[\mu_p\left(E_\gamma\right),\frac{4v_p}{s_p}\right] \qquad (2)$$

where:

Cp - turnover coefficient is equal to 1.6 · 10^{-13}(Wt · sec/MeV); $\varphi_\gamma\left(r_p,E_\gamma,t\right)$ - flux density of primary γ-radiation; $\mu_p\left(E_\gamma\right)$ - absorption coefficient with E_γ; E_γ - energy in body; ρ_p - body density; v_p - body volume; s_p - body surface; l_p - body length; t - time.

Flux density of primary γ-radiation is considered in-core energy release in the following:

$$\int_t\varphi_\gamma\left(r_p,E_\gamma,t\right)dt=f\left(E_\gamma,L_p\right),\mu_\Sigma\left(E_\gamma\right)^{-1}\left\{\xi_f\left(r_p,t\right)H_{inst}\left(E_\gamma\right)+\int_0^t\xi_f\left(r_p,\tau\right)H_{del}\left(E_\gamma,t-\tau\right)dt\right\} \qquad (3)$$

$$f\left(E_\gamma,L_p\right)=\exp\left[-\mu_p\left(E_\gamma\right)\rho_pL_p\right]G\left[\mu_\Sigma\left(E_\gamma\right)\cdot H,\mu_\Sigma\left(E_\gamma\right)R\right] \qquad (4)$$

$$G\left[\mu_\Sigma\left(E_\gamma\right)\cdot H,\mu_\Sigma\left(E_\gamma\right)R\right]=\int_0^H dz\int_0^R\exp\left[-\mu_\Sigma\left(E_\gamma\right)\cdot r\cdot\sec\Theta\right]\frac{r_pdr_p}{r_p^{\,2}+z^2} \qquad (5)$$

$$\sec\Theta=\frac{\sqrt{\left(r_p^{\,2}+r_0^{\,2}\right)}}{z} \qquad (6)$$

where:

$H_{inst}\left(E_\gamma\right)$ - instant γ-radiation spectra; $H_{del}\left(E_\gamma\right)$ - delayed γ-radiation spectra of fission products; $f\left(E_\gamma,L_p\right)$ - flux decrease function of γ-radiation in body; $\xi_f\left(r_p,t\right)$ - fission number per volume quantity of fuel element; $\xi_f\left(r_p,\tau\right)$ - fission number per volume of fuel element, regarding time >10 sec; $\mu_\Sigma\left(E_\gamma\right)$ - linear absorption of γ-quant's in reactor core area; H, R - cylinder height and radius; z, r_p - coordinates.

By applying partition and regarding leak function

$$f\left[\mu_p\left(E_\gamma\right), \frac{4v_p}{s_p}\right]$$

dependence of γ-quant's energy rate the following equation (1) may be derived accounting formula (3);

$$q_\gamma\left(r_p, E_\gamma, t\right) = \left\{Cp \cdot \bar{\mu}_p \cdot \rho_p \cdot f\left(\bar{\mu}_p, \frac{4v_p}{s_p}\right) \cdot \mu_\Sigma^{-1} \cdot \exp\left(-\bar{\mu}_p \cdot s_p L_p\right) \int_o^H dz \int_o^R \exp\left(-\bar{\mu}_\Sigma r_p \sec\Theta\right) \times \right.$$
$$\left. \times \frac{r_p dr_p}{r_p^{\;2} + z^2}\right\} \cdot \left[\int_{E_\gamma} \xi_f\left(r_p, t\right) H_{inst}\left(E_\gamma\right) E_\gamma dE_\gamma + \int_{E_\gamma}\int_0 \xi_f\left(r_p, t\right) H_{del}\left(E_\gamma, t-\tau\right) E_\gamma dE_\gamma d\tau\right] \quad (7)$$

Define the term in figure brackets as M_p and expression $I_{inst}\left(E_\gamma\right)$ and $I_{del}\left(E_\gamma, t-\tau\right)$ then approximating delayed γ-radiation as overall exponent sum:

$$q_\gamma\left(r_p, E_\gamma, t\right) = M_p\left\{\xi_f\left(r_p, t\right) I_{inst}\left(E_\gamma\right) \int_0^t \xi_f\left(r_p, t-\tau\right) \cdot \sum_n^N A(\beta)_m \exp\left[-\lambda(\beta)(t-\tau)_m\right] d\tau\right\} \quad (8)$$

where:

$A(\beta)$ - group effectiveness of fission products; $\lambda(\beta)$ - fission products decay constant.

For calculation $\xi_f\left(r_p, t\right)$ resolving in formula (8) Voters equation is used considering iteration Piker's method e.g.

$$\begin{cases} \xi_f\left(r_p, 0\right) = q_\gamma\left(r_p, E_\gamma, 0\right); \\ \xi_f\left(rt_p, t_1\right) = q_\gamma\left(r_p, E_\gamma, t_1\right) - M_p^{\;-1} I_{inst}\left(E_\gamma\right)^{-1} \times \\ \times \int_0^t \xi_f\left(r_p, 0\right) \sum_{k=1}^M A(\beta)_k \exp\left[-\lambda(\beta)\left(t_2 - \tau\right)\right]_k d\tau; \\ \vdots \\ \xi_f\left(rt_p, t_j\right) = q_\gamma\left(r_p, E_\gamma, t_j\right) - M_p^{\;-1} I_{inst}\left(E_\gamma\right)^{-1} \times \\ \times \int_0^t \xi_f\left(r_p, t_j\right) \sum_{k=1}^M A(\beta)_k \exp\left[-\lambda(\beta)\left(t_2 - \tau\right)_k\right]_k d\tau; \end{cases} \quad (9)$$
$$j = 1, 2, \ldots i$$

In stationary pile mode operation dose rate absorbed in cylinder body may be determined by formula:

$$q_\gamma\left(r_p, E_\gamma\right) = \int_{E_\gamma} M_p\left(E_\gamma\right) \xi_f H_{inst}\left(E_\gamma\right) E_\gamma dE_\gamma \quad (10)$$

For this case the problem of in-core γ-spectra determination will be reduced to resolution of system of integral equations.

Using recurrent relations we obtain:

$$H_{inst}\left(E_{\gamma}\right)_{\gamma}^{n+1}=8\exp\left(1{,}1E\gamma\right)\prod_{i=1}^{k}\left[\sum_{j=1}^{?}\frac{M_{p}\left(E_{\gamma}\right)_{ij}}{M_{p}\left(E_{\gamma}\right)_{ij}H_{inst}\left(E_{\gamma}\right)_{j}^{n}}\div\sum_{j=1}^{l}\frac{M_{p}\left(E_{\gamma}\right)_{ij}}{q_{y}\left(r_{p},E_{\gamma}\right)_{i}}\right] \quad (11)$$

where:

$H_{inst}\left(E_{\gamma}\right)_{j}^{n+1}$ - instant γ-specter in γ-energy group with n+1 iteration; i - energy gap number (group).

Final trial for iteration process quit determination is the resolution of the following assumption:

$$\max\left[1-q_{\gamma}\left(r_{p},E_{\gamma}\right)_{i}\Big/q_{\gamma}\left(r_{p},E_{\gamma}\right)^{n}\right]<\xi_{0} \quad (12)$$

$$\xi_{0}=\delta\xi\sum_{m=1}^{M}Sq^{2}\left[q_{\gamma}\left(r_{p},E_{\gamma}\right)\right]_{m}^{-1} \quad (13)$$

where:

ξ_0 - assigned accuracy;

$\delta\xi$ - factor determined from test γ-specter reduction;

Sq - relative dispersion of power shift in cylinder body.

Consequently whilst inserting in-core assigned body numbers fabricated from materials with various atomic number and absorption factors having insignificant contributions into elastic and inelastic fast neutron scattering and radiation capture e.g. (Bi, Pb, Zr, Fe, W) and dose rate determination energy in-core release may be determined with a reliable accuracy (± 20%) for practical purpose.

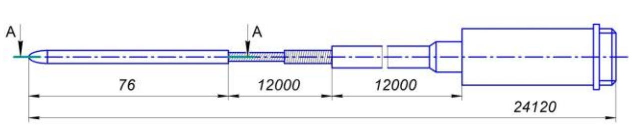

Figure 5.Calorimetric γ- sensor.

For energy release control handing application γ-calorimeter design has been developed having performance body sensitive and grade scale elements [29-33]. Calorimetric sensor (See fig. 5) subjected full scale performance testing in operating nuclear reactors. Experiments carried out with Gamma calorimeter proved that primary γ-radiation contribution for WWER accounted 80% and secondary - 20% respectively.

The share instant γ-radiation accounted almost 70% and delayed 30%. In Fig. 6 relative axial power distributions are presented (φ_z) as measured by gamma-calorimeter (1) and activation detector (2) and profile BIPR-5 calculated for WWER-440.

Carried out tests on control rod insertions showed that 2/3 of primary γ-radiation was absorbed in a fuel element assembly where GC was located and γ-radiation share from surrounding fuel elements in overall dose rate estimation overcome 15%, being in agreement with assumption states before.

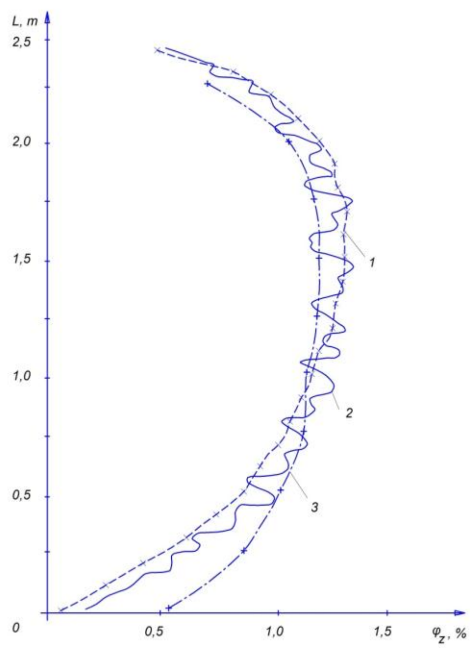

Figure 6. Relative power distributions

5. Conclusions

Nuclear Power Plants energy production is closely related to reliability and efficiency of plant operation, engineering characteristics promotion, in core geometry dimension and energy stress increase, thereby a considerable buildup in method and means of in core reactor control is observed, following sophisticated IRCS function performance (diagnostic, identification, anticipated operational occurrences). Therefore the most important problems facing new types of IRCS design need further investigation and shape the following:

- further research in non-conventional control methods application;
- selection of available and reliable in core instrumentation excluding sensitive element burn up with availability of self-calibration of premising in core detectors;
- optimization of most informative fluxes and features in anticipated operational occurrence on the early stage of their development;
- effective procedure formation for treatment and analysis of measurement information and recognition modes for accident condition;
- development of mathematical base and software for realization of above procedures;
- determination of makeup and architecture of information management complex, detector sets, cable trains, software regarding NPP type;
- further investigation in human-computer interface.

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