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Determination of Research Reactor Safety Parameters by Reactor Calculations

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DETERMINATION OF RESEARCH REACTOR SAFETY PARAMETERS BY REACTOR CALCULATIONS

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Main research reactor safety parameters such as power density peaking factors, shutdown margin and temperature reactivity coefficients are treated. Reactor physics explanation of the parameters is given together with their application in safety evaluation performed as part of research reactor operation. Reactor calculations are presented as a method for their determination assuming use of widely available computer codes.

1 Introduction

Several activities related to normal research reactor operation involve safety evaluation. In principle, any activity that may influence neutronic, thermal-hydraulic and mechanical properties of the reactor should be supported by safety evaluation. Typical activities involving safety evaluation are as follows:

modifications of reactor components,
power uprating,
change of fuel element type,
mixed core operation,
new experimental setups,
pulsing,
spent fuel storing,
aging, etc.

Complete safety evaluation related to major changes in reactor system and resulting in safety analysis report modifications is normally performed by competent institutions (e.g. reactor manufacturer, national institutes, consultant companies). Research reactor operation team is normally neither qualified nor equipped for such work particularly in small research reactor centers. However, due to experimental nature of research reactors certain activities requiring safety evaluations should be routinely performed by the reactor operation team (e.g. modifications of core configuration due to experiments, fuel management). Purpose of this paper to explain the most important reactor physics safety parameters of a small research reactor from the aspect of reactor operator. The following safety related quantities and parameters are treated:

power distributions and power density peaking,
shutdown margin, control rod worth and excess reactivity and

temperature reactivity coefficients.

Use of widely available computer codes and integrated packages adapted to the practical needs in the core management is explained.

2 Research reactor safety parameters

General aspects of research reactor calculations are treated in ref. [1]. Basic principles and physical models are presented for small TRIGA and plate type MTR reactors together with widely available core management computer codes for their practical calculations. Ref. [1] is recommended as additional material to this presentation.

2.1 Power distributions

Fuel temperature is one of the most important limiting conditions on reactor operation. It depends on the reactor design, thermal-hydraulics properties and on the power density released in fuel. Power density distribution depends on core configuration and loading pattern. It changes due to routine core and fuel management: transition from initial fresh core to normal operation core (adding fuel elements), spent fuel replacement, modifications of loading pattern, mixed core operation, introduction of in-core irradiation channels, etc. It is also affected by the burn-up even if the loading pattern is not changed. It is the responsibility of the reactor operator to keep the fuel temperature and consequently the maximum power density within the limits prescribed by the safety analysis report.

The limitations are normally explicitly imposed on maximum fuel temperature. The limiting temperatures and other thermal hydraulic parameters are defined by the design properties of reactor fuel and other components (mechanical design, cladding stress, corrosion). For example, the maximum temperature in standard TRIGA fuel rod is limited to $\approx 1000^0$ C by the internal pressure due to dissociation of Hydrogen in Zirconium hydride at high temperature. As the temperature is in steady state conditions approximately proportional to power the temperature limitations implicitly define also the power density limitations (Fig.1).

Power density limitations are result of the thermal-hydraulics analysis that is normally beyond the reach of reactor operation team. The limitations are derived from the assumptions on the thermal power density distribution and its integral equal to total reactor power. The assumptions on power density distribution are normally conservative covering a wide range of modes and conditions in reactor operation. In principle the assumptions on power distributions are included in the safety analysis report as they make input to the thermal-hydraulics analysis. It is the responsibility of the reactor operator to accommodate the operating conditions, loading pattern and the experiments such that the power limitations are not

violated. However, this requirement is very difficult to respect due to practical reasons: experimental information on power distributions in the fuel is very limited. The only on-line information is normally fuel temperature. Power distributions are measured only periodically by measuring flux distribution. However, in rod-type reactors such measurements yield only indirect information on flux and power inside fuel elements which is the actual limitation. For this reason, reactor calculations are the most practical and feasible method for power density distribution analysis. However, their users must be well aware of the physical models, their accuracy and limitations before using their results in practice [2].

The power distribution in a research reactor depends on several conditions: fuel type and enrichment, loading pattern (asymmetric and mixed cores), in-core water gaps, axial and radial influence of (partly) inserted control rods, burn-up, etc. Research reactor cores are normally small, of irregular shape and heterogeneous. Power density distribution is peaked, tilted and complicated. Example of a typical flux and power distribution in a small mixed TRIGA reactor is presented in Figs. 2-4. Note that the maximum power density occurs near water gaps (empty positions) due to increased thermal flux.

It is important to note that the power density depends also on the volume of the core. In research reactors it is normal that the effective volume of the core is changed because the number of fuel elements in the reactor is not fixed. Example: 1MW TRIGA reactor starts operation with ≈50 fresh fuel elements. During operation fresh fuel elements are added to compensate excess reactivity reduction due to burn-up. Equilibrium operation core contains more than 100 fuel elements. Even if it is assumed that relative power distribution is not changed the average and maximum fuel rod power are decreased for 50%.

Power distribution in a bare, homogeneous cylindrical reactor can be calculated in analytical way. In radial direction it is proportional to Bessel function $J_0(r)$ and in axial to cos(z) with peak-to-average values ≈ 2 and ≈1.6, respectively (note: maximum power density is ≈3.2 times higher than the average). Real research reactor cores are always reflected (water, graphite, beryllium) and their peak-to-average radial and axial power density values are reduced (Figs. 5 and 6) typically to 1.7 and 1.4, respectively (these are the values normally assumed in thermal-hydraulics analysis in small TRIGA reactors). Radial heterogeneity may, however, significantly increase local radial power peaking values (the axial power peaking factor is not so sensitive because the axial structure of the reactor is normally not changed).

The effect of core heterogeneity on power distributions is elaborated in details in ref. [3] enclosed to this presentation as additional material. Effects of rod internal power distribution are described as well.

Power distribution calculations are normally performed in diffusion approximation assuming unit cell homogenization (example of such codes for TRIGA reactor calculations is TRIGLAV [4]). Each fuel rod is homogenized with water and other components of the unit cell (explicit modeling of fuel rods is feasible only in Monte-Carlo calculations). Also the power density distribution is smeared over the entire unit-cell volume and has no physical meaning in a particular point. Only its integral over the unit cell corresponds to a quantity that has physical meaning. It is equal to the fuel rod power (per definition). The result of the diffusion calculation is fuel rod power distribution for all elements in the core. Maximum rod power can be determined as it is normally one of the limiting values in the safety analysis.

In most cases the maximum rod power determines maximum fuel temperature and other thermal-hydraulics and mechanical design limiting parameters. However, in cores with large power gradients between the fuel elements (e.g. mixed cores, vicinity of water gaps and irradiation channels, effect of control rods and control blades in plate type reactors) the power density *inside* the fuel element must be determined for temperature calculation as well. The diffusion calculation results can not be used as they apply to smeared unit cells. The power distribution in the fuel rod is normally calculated in unit-cell or multi-cell approximation using a transport or Monte-Carlo code capable of detailed fuel rod modeling. The fuel element must be modeled together with its first (or even second) neighbors in order to reproduce the flux gradients realistically (optimum is evidently modeling of entire core which is normally not feasible with transport or Monte-Carlo code). WIMS [5] is an example of codes that may be used in fuel rod radial and axial power distribution calculation. Experience shows that there are no essential differences in quality of results between old WIMS-D/4 and new WIMS-D/5 version for simple models used in research reactor calculation. One dimensional super-cell approximation is sufficient in some cases (e.g. mixed core effects), however two-dimensional approximation has to be applied in general.

Relative accuracy of rod power calculation is in the order of 0.15 (see Fig. 5). Main sources of inaccuracy are:

- simplifications in physical model (power distribution tilts due to inappropriate modeling of leakage, inappropriate modeling of unit-cell neighbors, diffusion coefficient determination)
- material and geometry uncertainties (initial isotopic composition, burn-up, fission products, fuel temperature, water density)
- uncertainties and errors in reactor power calibration (power redistribution effects due to localized power detectors).

The uncertainties are elaborated in details in refs. [6,7,8].

2.2. Shutdown margin

The shutdown margin is normally defined as negative reactivity by which the reactor is subcritical if all control rods were fully inserted in the core except the most reactive one. By this requirement it is provided that the reactor can be made subcritical even if one of the control rods fails (the one with the highest worth which is for this reason usually called the safety rod). The minimum limiting value for the shutdown margin is zero: in principle it is sufficient to be able to shutdown the reactor. However, normally the reactor designers and licensing authorities increase the limiting value for some conservatism taking into account uncertainties in control rod worth and excess reactivity determination. Typical shutdown margin value for research reactors is 0.5$ ($1\$=1\beta_{eff}=0.007$).

Shutdown margin is normally determined in experimental way by measuring the excess reactivity and the worth of all control rods at zero power xenon free conditions. It is recommended to measure it every time when the core configuration is changed as the control rod worth and excess reactivity both depend on the number and type of fuel elements in the core, their burn-up and loading pattern.

Research reactor fuel management codes based on diffusion approximation for the core and simple (multi) cell transport model for the cell are not appropriate for control rod worth calculation in research reactors without experimental adjustments. Experience shows that the error of such calculation without empirical adjustments of the models is in the order of 50%. Empirical adjustments can be simply introduced by changing the effective radius (surface) of the absorber material as it is black for thermal neutrons. However, control rod worth calculations are normally not performed with simple fuel management codes normally available to the reactor operator team as they require more sophisticated design level codes.

The shutdown margin is equal excess reactivity minus sum of all control rod worth except the most reactive one. In contrast to the control rod worth the excess reactivity may be quite accurately calculated and predicted using core management codes. The absolute accuracy in k-eff of two-dimensional diffusion codes is in the order of 1-2% mainly due to systematic errors in modeling radial and axial leakage (note large contribution of leakage to the neutron balance in small research reactors; typically unit cell k-inf=1.4). The relative accuracy in criticality calculation is much better. The accuracy of Δk-eff calculation is in the order of 10^{-3} even if the differences in core size and structure are significant provided that the calculations apply to the same reactor.

Adjusting the axial buckling can compensate the systematic errors in two-dimensional k-eff calculation. It is convenient to adjust the buckling to the first core experimental critical configuration when the fuel is fresh and excess reactivity

is well defined. Besides leakage such adjustment covers also other systematic uncertainties and errors (e.g. material composition) except reactivity effects of loading pattern changes, burn-up, power defect and xenon poisoning. Excess reactivity changes due to these effects can be calculated with better accuracy not influenced by the systematic errors due to simplified geometry and leakage models.

The accuracy of burn-up calculations depends mainly on the isotopic burn-up schemes and data used in the depletion code (e.g. WIMS). The capture and fission in U-235 and absorption in U-238 and fission products are strongly predominant reactions. Production of Pu isotopes by resonance absorption in U-238 and their fission are in research reactors not important due to high enrichment and uranium concentration. Consequently spectrum deformations and shifts due to burn-up do not influence isotopic composition changes like in low enriched power reactors. The excess reactivity changes with burn-up are in research reactors mainly sensitive to the uranium, main fission product and in particular cases burnable poison nuclear data (e.g. Er in TRIGA fuel).

Comparison of calculated burn-up reactivity reduction slopes to the experimental ones often shows much bigger discrepancy than expected from the quality of nuclear data affecting burn-up and other inaccuracies of the calculation models [8]. One of the most frequent reasons is systematic error in power calibration as the burn-up is proportional to reactor power. In small research reactors the power is normally calibrated with respect to a single neutron detector. Its response is proportional to the flux at its position. Local flux is proportional to the total flux (power) of the reactor only if its radial and axial distributions do not change. This is however not the case in the research reactors where operational reactivity changes (burn-up, power defect, xenon effect) are compensated by moving the control rods. Redistribution effects on neutron flux distribution due to control rod insertion/withdrawal detected by a single detector may be in the order of 20% yielding the same error in reactor power readings. Using two or more detectors strategically located at different locations around the core can reduce the error [9].

2.3. Temperature reactivity coefficients and power defect

Temperature reactivity coefficient is defined as reactivity change per unit temperature change. Three temperature coefficients are normally defined with respect to which temperature change is considered: fuel temperature reactivity coefficient (also in highly enriched research reactors sometimes mistakenly denoted as Doppler coefficient), coolant temperature reactivity coefficient (sometimes denoted moderator temperature reactivity coefficient also in TRIGA reactors where principal moderator is included in the fuel material) and isothermal reactivity coefficient. Fuel temperature coefficient is defined as reactivity change

per unit fuel temperature change at fixed coolant temperature. Coolant temperature coefficient is defined as reactivity change per unit coolant temperature change at fixed fuel temperature. Isothermal coefficient is defined as reactivity change per unit change of fuel and coolant temperature. If moderator is different from the coolant (e.g. heavy water reactors) the moderator reactivity coefficient can be defined as well.

Fuel temperature reactivity coefficient is important for reactivity and power excursion transient analysis where power feedback effects depend on the sign, rate and time delay of fuel temperature reactivity effects. Negative and possibly prompt fuel temperature reactivity coefficient is one of basic safety requirements in most research reactors. Main contribution to the coolant temperature coefficient is water density temperature variation. It can be easily related to the void reactivity coefficient. It is negative in undermoderated and positive in overmoderated reactors. It is important in reactivity analysis of coolant flow and temperature (e.g. blocked coolant channel, flow reversal). Isothermal coefficient is important because it is the only one that can be accurately measured.

The calculation of temperature coefficients is feasible only for special or hypothetical conditions. In calculations it is easy to change fuel temperature without changing coolant temperature, however, this is very difficult to carry out in practical experiment. Moreover, fuel temperature at power does not change uniformly in all fuel elements. In the calculation and in the measurement it is necessary to consider the radial and axial temperature distribution.

The core management codes are not appropriate for temperature reactivity coefficient calculations without modifications. However, experience shows that good estimates of the coefficients can be obtained in unit cell approximation (e.g. WIMS) provided that the core is uniform (Fig. 8).

This is not true in mixed cores. The coefficients are not simple mixture of the unit-cell coefficients of the constituents. The contributions of different fuel elements depend on their neutron importance. At least two-dimensional full-core models are required for reliable estimates.

Fuel temperature and power are related. Power coefficient (defined as reactivity change per unit reactor power) and defect (power defect is integral of power coefficient from zero power to a certain power) are easily measured. Core management codes are normally designed to reproduce the power defect properly by self-adjusting temperature and power distributions using empirical correlation between fuel element power and temperature. The effect of temperature/power self-adjusting is in research reactors not as strong as in big power reactors where power distributions are 'soft' and sensitive to small perturbations.

The temperature reactivity coefficient calculations are described in details in ref. [3] that is recommended as additional material to this presentation.

3 Conclusions

Some general aspects of safety related research reactor parameter calculations are discussed. Their specific aspects depend on the reactor and fuel type, operating conditions and software available for reactor calculations. The computer codes and the calculation procedures must be well understood and verified before being applied to practical core safety analysis in particular situation.

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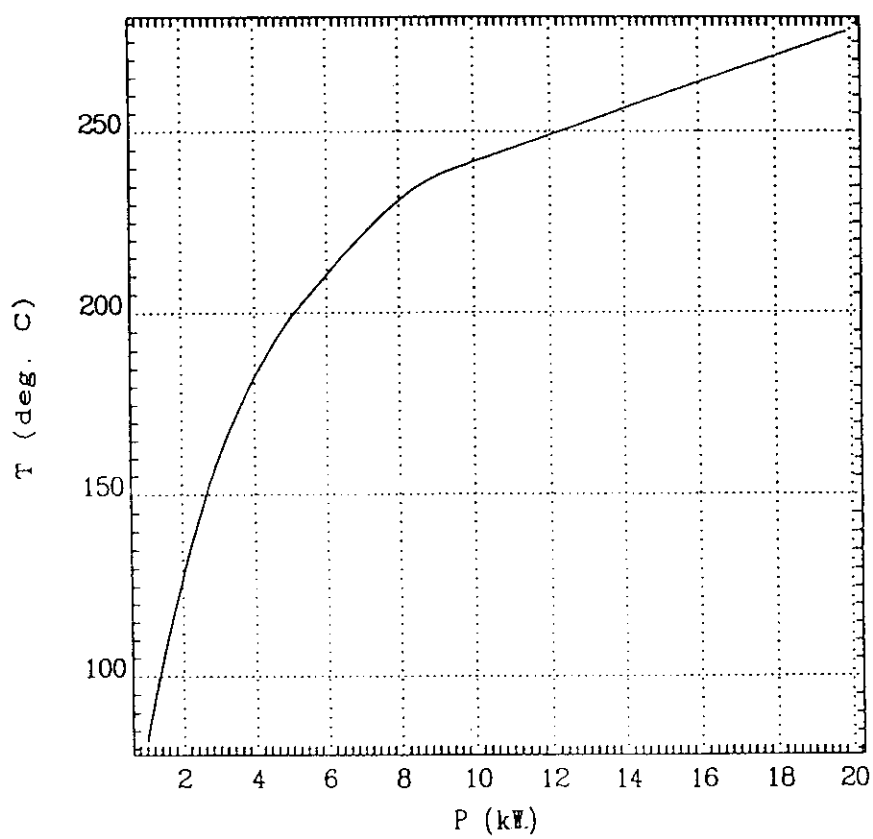

Figure 1: Typical relation between rod power and average fuel temperature for standard TRIGA fuel rod in 1MW reactor

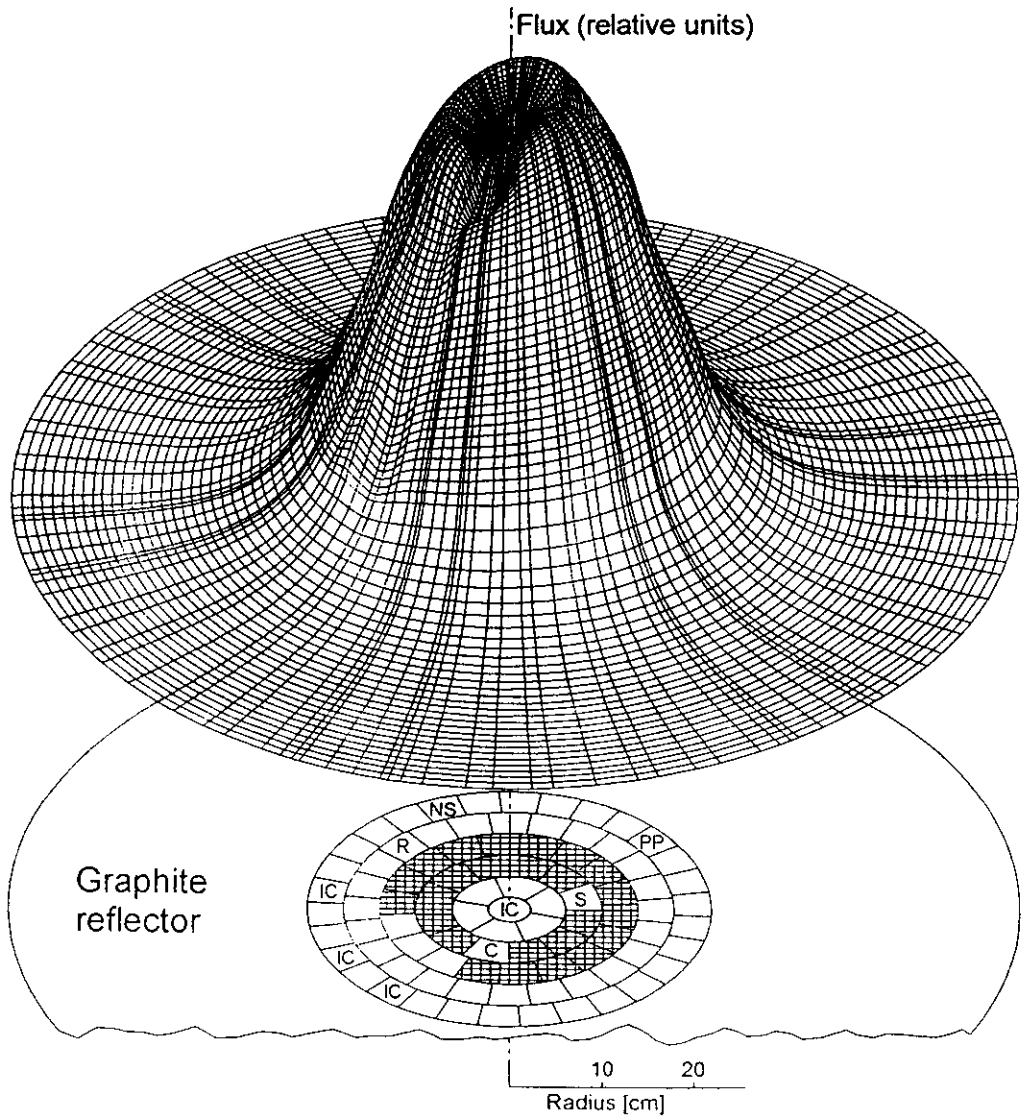

Figure 2: Fast flux (>1eV) radial distribution in a typical mixed TRIGA reactor, calculated with TRIGLAV code (dark core positions: 70% enriched FLIP fuel, white core positions: 20% enriched standard fuel, both 8% U-content, S,C,R denote control rod positions, IC,PP irradiation channels, NS neutron source)

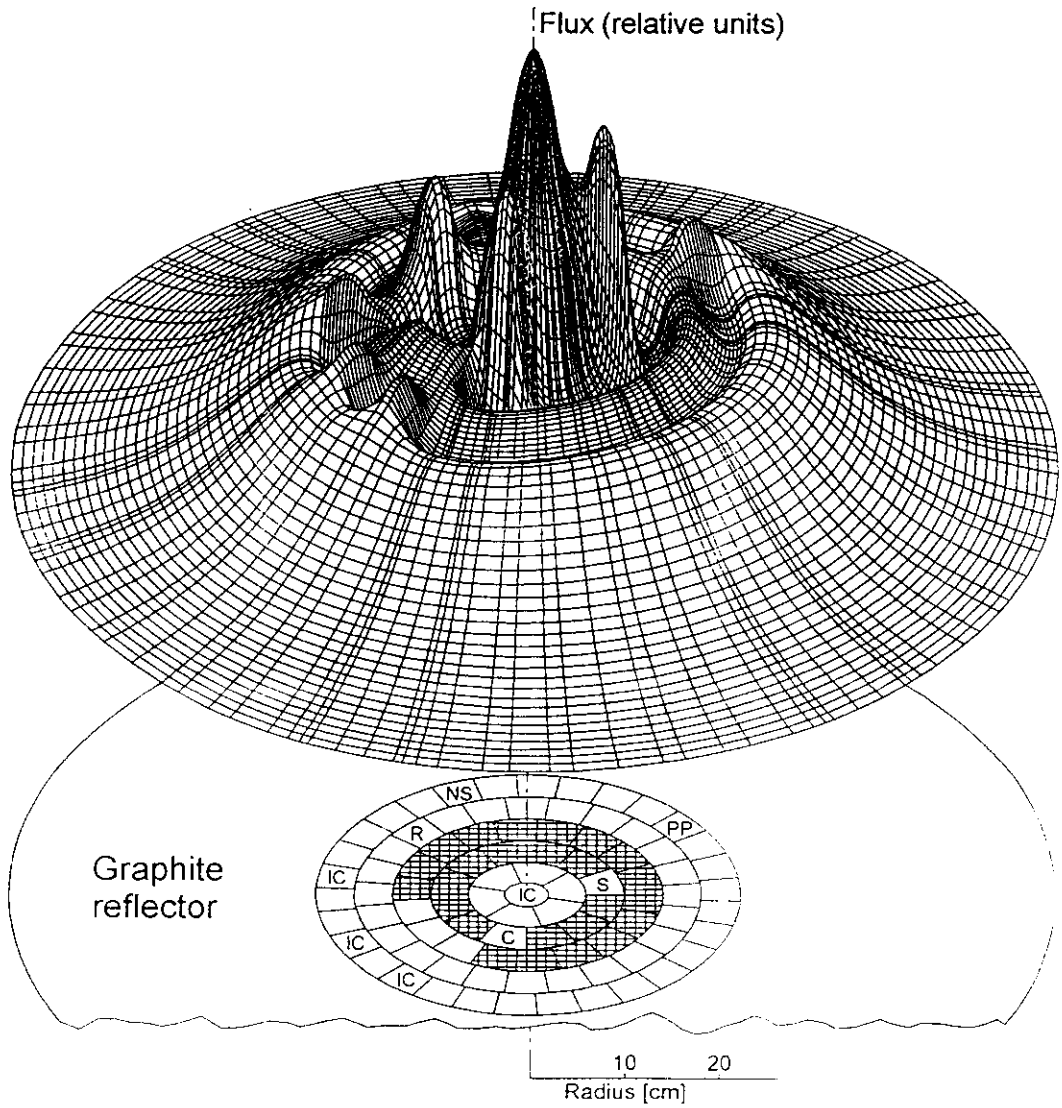

Figure 3: Thermal flux (<1eV) radial distribution in a typical mixed TRIGA reactor, calculated with TRIGLAV code (dark core positions: 70% enriched FLIP fuel, white core positions: 20% enriched standard fuel, both 8% U-content, S,C,R denote control rod positions, IC,PP irradiation channels, NS neutron source)

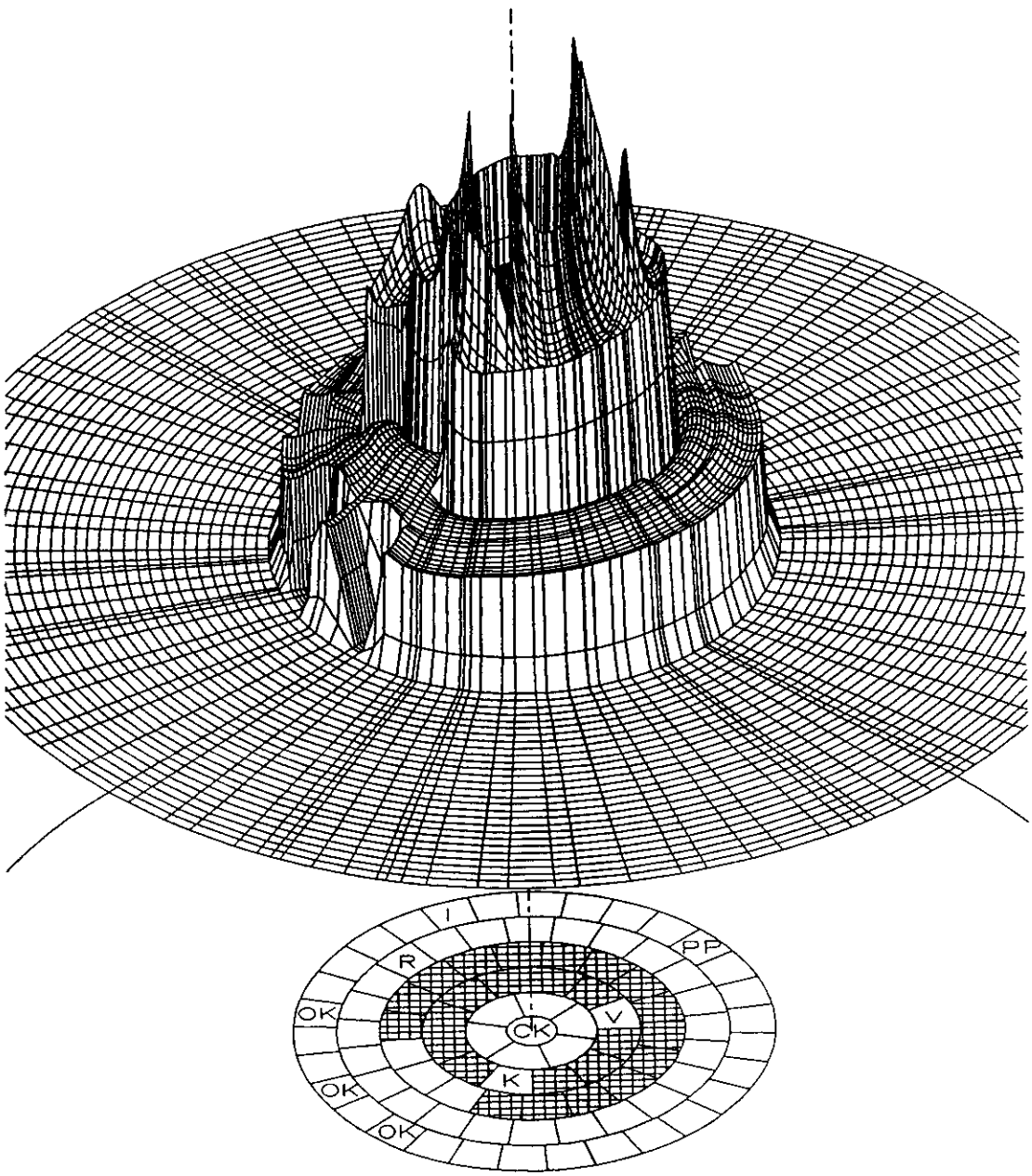

Figure 4: Power density radial distribution in a typical mixed TRIGA core (same configuration as in Figs. 2 and 3), calculated with TRIGLAV code (dark core positions: 70% enriched FLIP fuel, white core positions: 20% enriched standard fuel, both 8% U-content, V,K,R denote control rod positions, OK,PP irradiation channels, I neutron source)

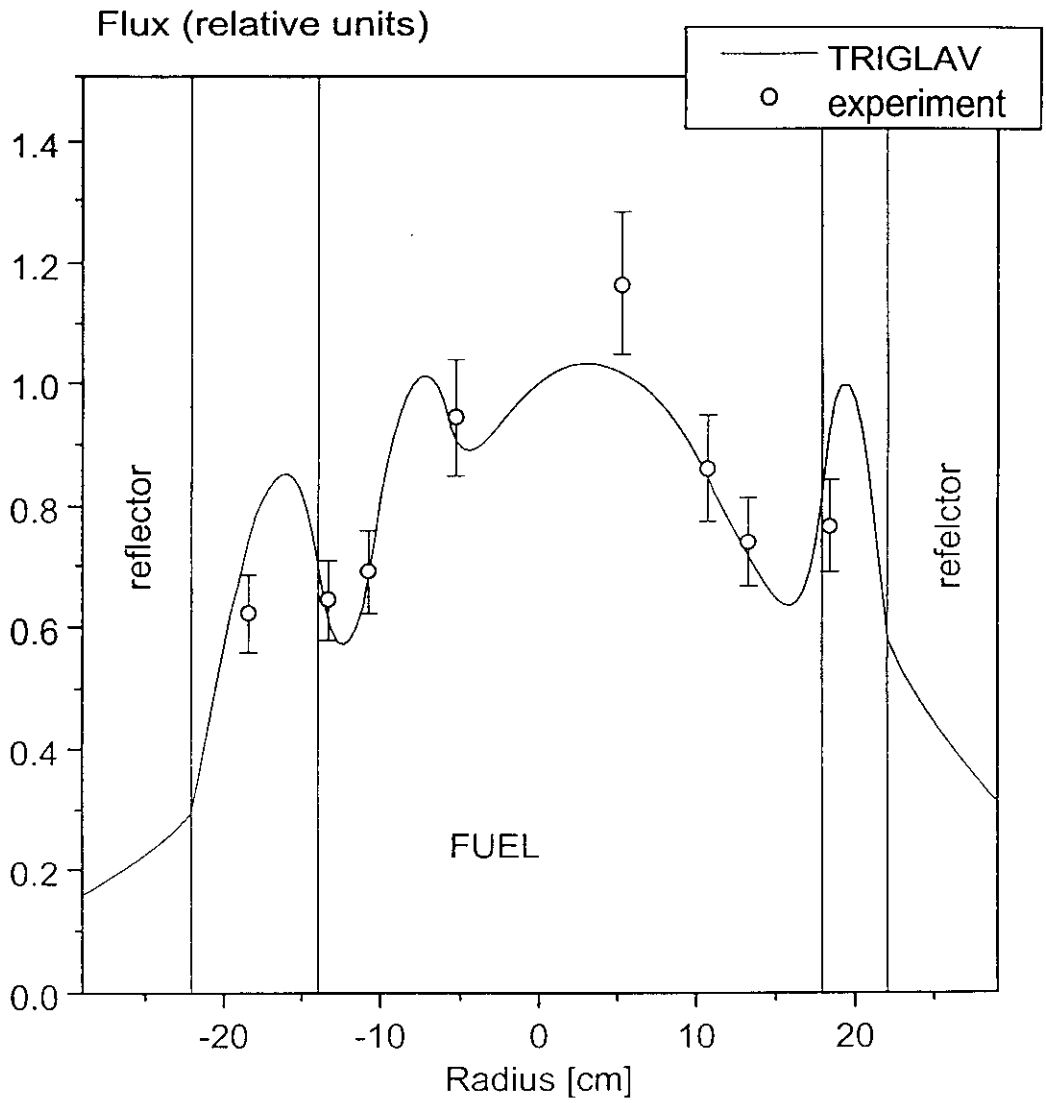

Figure 5: Measured and calculated radial flux distribution for fresh uniform TRIGA graphite reflected core (12% U-content 20% enriched fresh Standard fuel only, benchmark core, ref. [7]).

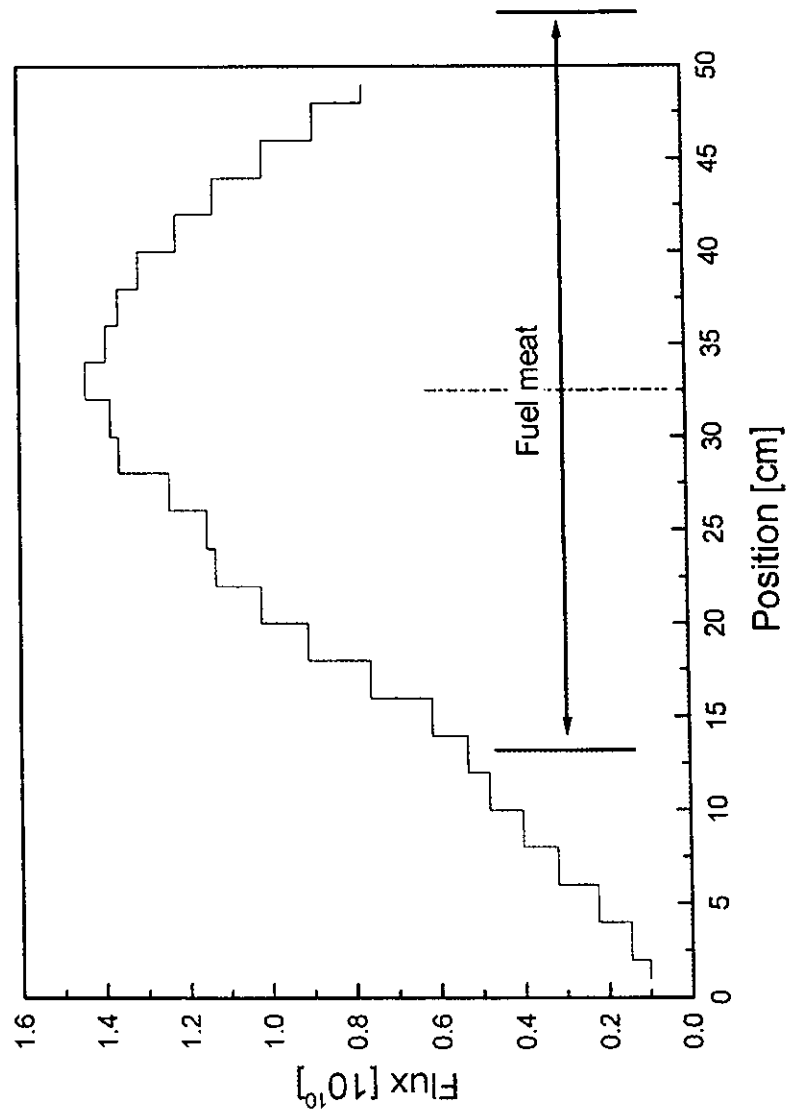

Figure 6: Radially averaged measured axial flux distribution for fresh uniform TRIGA graphite reflected core (12% U-content 20% enriched fresh Standard fuel only, benchmark core, ref. [7]). Measurement performed with copper wire in 15 radial grid-plate positions, 0 on abscissa is at top of upper grid-plate.

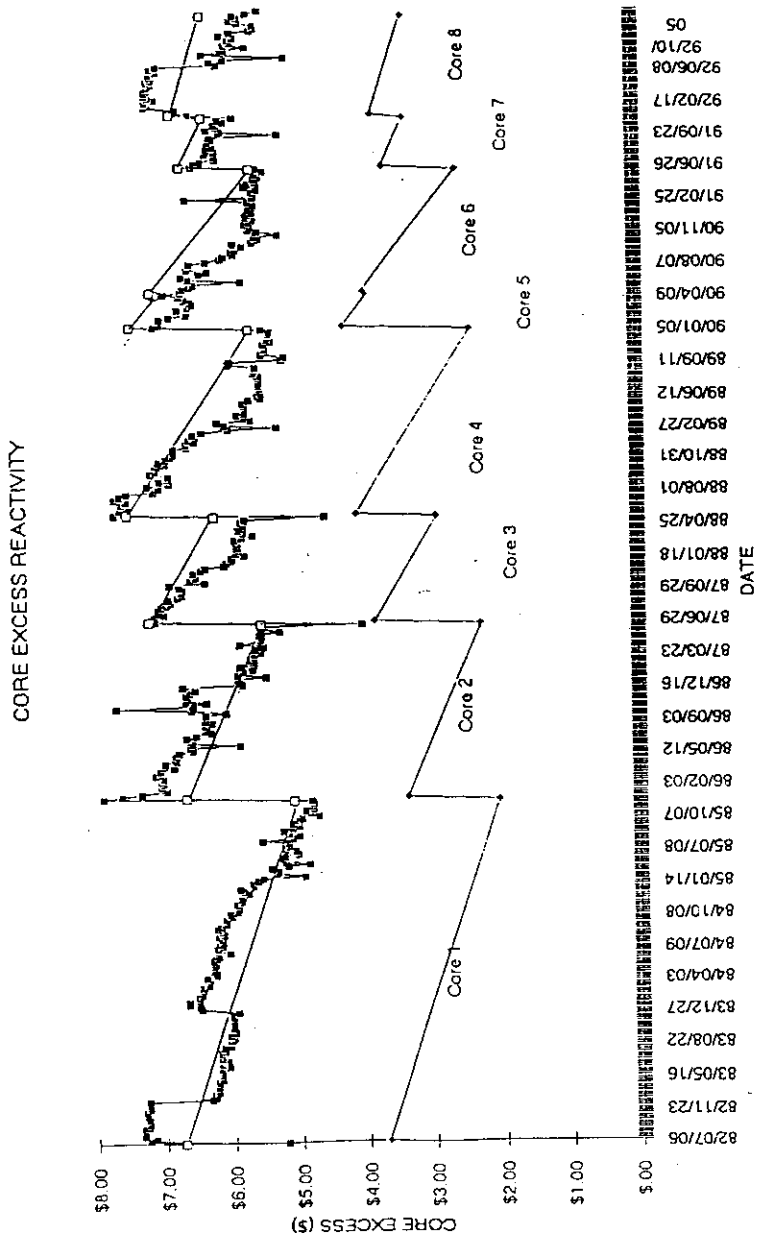

Figure 7. Example of core excess reactivity calculation in comparison to measurements for long term operation, 1MW TRIGA reactor. Calculations performed for zero and full power xenon free conditions.

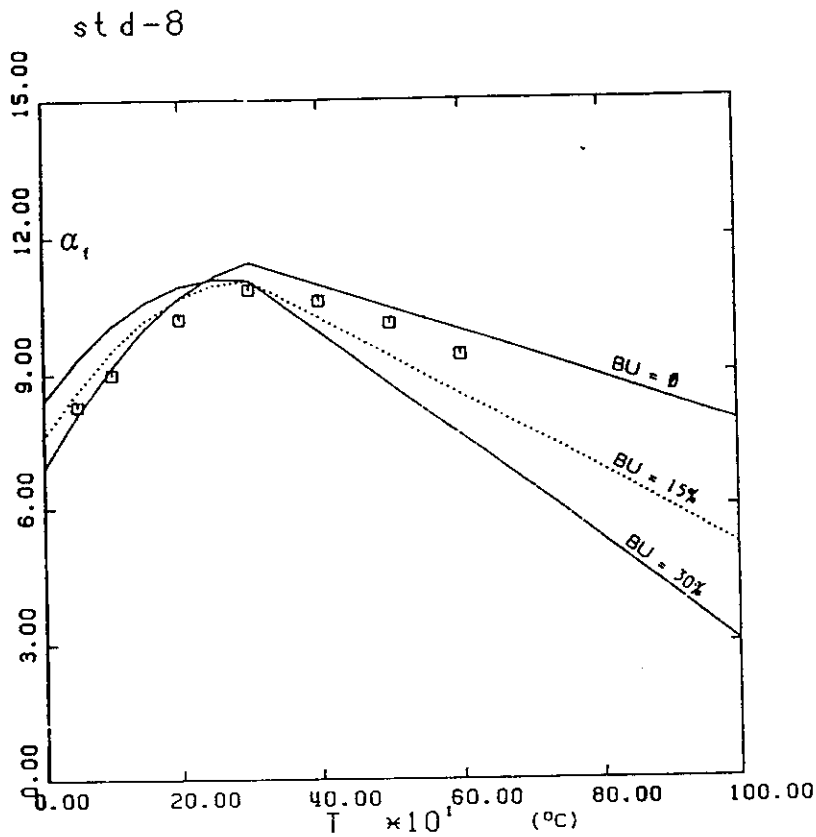

Figure 8. Standard TRIGA fuel temperature coefficient α_f (calculated with WIMS in unit cell approximation compared to design values. Units of α_f are $10^{-5}\delta k/k/^0C$