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THE EFFECT OF THE VOLUMETRIC HEAT SOURCE DISTRIBUTION OF THE FUEL PELLET ON THE MINIMUM DNBR RATIO

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ABSTRACT - The radial power distribution in a VVER-440 type fuel assembly is strongly non-uniform as a result of the water-gap between the shrouds and the moderator filled central tube. Consequently, it can be expected that the power density inside a single fuel rod is inhomogeneous, as well.
In the paper the methodology and the results of coupled thermohydraulic and neutronic calculations are presented. The objective of the analysis was the investigation of the heat source distribution and the determination of the possible extent of the power non-uniformity in a corner rod which has always the highest peaking factor in a VVER-440 type assembly.
The results of the analysis revealed that there can be a strong non-uniformity of power distribution inside a fuel pellet, and the effect depends first of all on the general assembly conditions, while the local subchannel parameters have only a slight influence on the pellet heat source distribution.

1. Introduction

It is of general practice that the safety analysis of a nuclear power plant must prove that in the course of Anticipated Operational Occurrences no fuel failure will occur. According to the acceptance criteria, cladding failure has to be supposed when the DNBR ratio becomes less than a minimum value defined for the given type of fuel assembly.

The analysis of DNBR ratio is often performed by a subchannel code like COBRA where the axial and radial power distributions in the investigated fuel assembly are taken from the calculation of core neutron physics. To obtain the minimum possible value of the DNBR, various kinds of parameter effects and uncertainties have to be taken into consideration, however, the non-uniformity of the volumetric heat source distribution inside a single fuel rod is generally neglected.

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In the paper the methodology and the results of coupled thermohydraulic and neutronic calculations are presented. The objective of the analysis was the investigation of the heat source distribution and the determination of the possible extent of the power non-uniformity in a corner rod which has always the highest peaking factor in a VVER-440 type assembly.

2. Definition and solution of the problem

The analysis of reactor transients where the key-safety-parameter is the DNBR, requires the application of a thermohydraulic system code calculating the general behaviour of the core and a detailed investigation of the conditions in the assembly containing the hot fuel rod. The first step can be accomplished by the RELAP code while the detailed analysis is generally carried out using the COBRA subchannel program. The power distribution for the thermohydraulic calculations is provided by the relevant reactor physics computer codes. In the practice of the safety analysis the calculation of the azimuthal heat source distribution in a single fuel rod and the change of the DNBR along the periphery of the hot fuel element are not required. The correct consideration of the effect of the volumetric heat source distribution needs the two-dimensional coupled calculation of the thermohydraulic and reactor physics parameters of a fuel pellet.

To solve this problem a 2D heat conduction code FUTE [1] and MGCP2D [2] have been developed. The FUTE code is enabled to calculate the cladding temperatures. The gap modelling takes into account gap conductance. The boundary condition was defined by means of the coolant heat transfer coefficient. It has been found that the method of collision probabilities is the most suitable model for the neutron physics calculationin in the MGCP2D code. The expressions for the collision probabilities were derived from the integral form of the transport equations. The CP code MGCP2D is described in Refs. [2] and [3].

The flow chart of the complex calculation is illustrated in Fig. 1.

The first step is the use of the KARATE-440 code [4] for the calculation of the fuel assembly containing the hot rod. In the calculation of the assembly the neighbouring assemblies are also considered. The resultsing quantities are as follows:

- neutron current in two energy groups at the corner cell boundaries,
- thermal hydraulic parameters of the assembly subchannels (coolant temperatures, heat transfer coefficients, etc.).

Using the MULTICELL module [5], the boundary conditions for the corner cell are generated in 70 groups. These data represent the input for the CP code.

As a module of the KARATE-440 code, COBRA is used for the calculation of the thermal hydraulic parameters. The resulting quantities are the subchannel data. For further use it is necessary to convert them into parameters characterizing the azimuthal subcells which form the basis of the heat source distribution calculation.

The next step is a two-dimensional fuel element temperature distribution calculation by means of the FUTE program developed for this purpose. Having the temperature distribution, all the input data are available to run the Collision Probability code MGCP2D.

The resulting quantity is the heat source distribution inside the fuel pellet. An iteration procedure is applied until the fuel power distribution is obtained with sufficient accuracy.

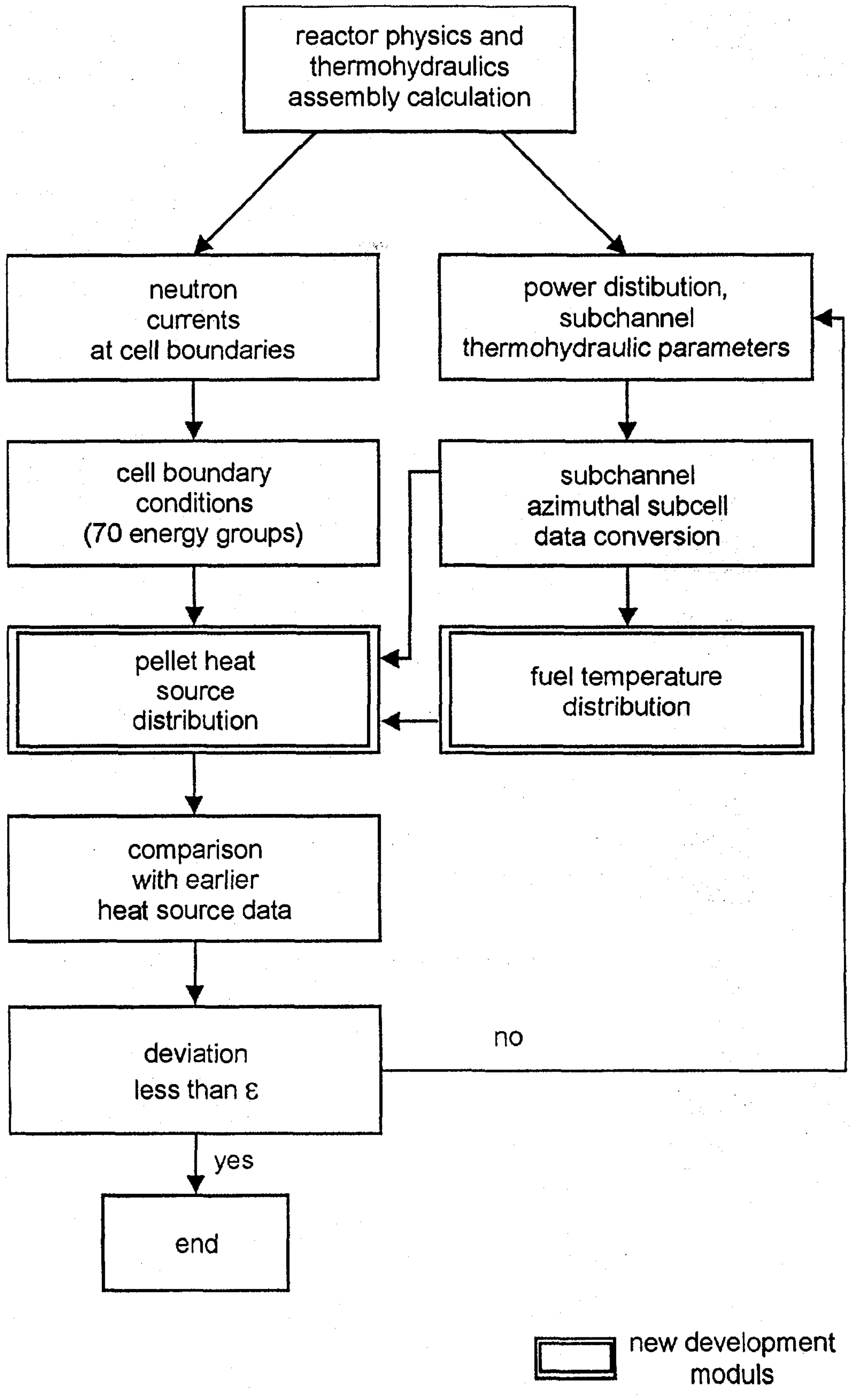


Fig.1 Flow chart of the calculation

3. Definition of the cases selected for investigation

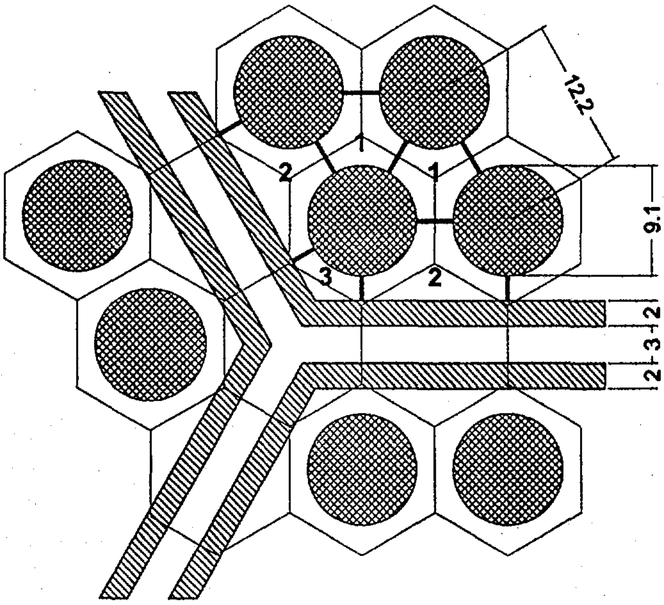
To assess the effect of non-uniform pellet heat source distribution, two extreme cases have been selected:

(A) Single phase case with subcooled outlet
(B) Two-phase case with high quality outlet.

The calculations have been performed for VVER-440 fuel assemblies of the Paks NPP considering the following general conditions:

- Enrichment in the selected assembly		3.6 %
- Enrichment in the neighbouring assemblies		1.6 %
- Boric acid concentration		5.2 g/kg
- Axial power distribution		uniform
- Inlet temperature		265.9 °C
- System pressure		123.6 bar
- Inlet mass flux		2590 kg/m^{2}s
- Average power density of the selected assembly	- Case A:	133.6 W/cm^3
	- Case B:	172.9 W/cm^3

The geometry of the corner part of the assembly is shown in Fig. 2, where the most important dimensions are also given in millimetres. In the figure - for the separation of subchannels and cells - thick and thin lines are used, respectively.



1 - normal; 2 - side; 3 - corner subchannels

Fig. 2. Nodalization of thermohydraulics and neutronphysics calculation

—— neutronphysics cells; —— thermohydraulics subchannels

4. Results

The results of the thermohydraulic calculations are summarized in Tables 1 and 2 (where the numbering of the subcells is shown in Fig. 3). In the first case when the coolant is in subcooled condition, the mass flux is high since the hydraulic resistances of the channels are relatively small. The single phase outlet means that the equilibrium quality is zero and the density values are high.

In the second case the outlet quality is rather high and consequently the coolant densities are moderate.

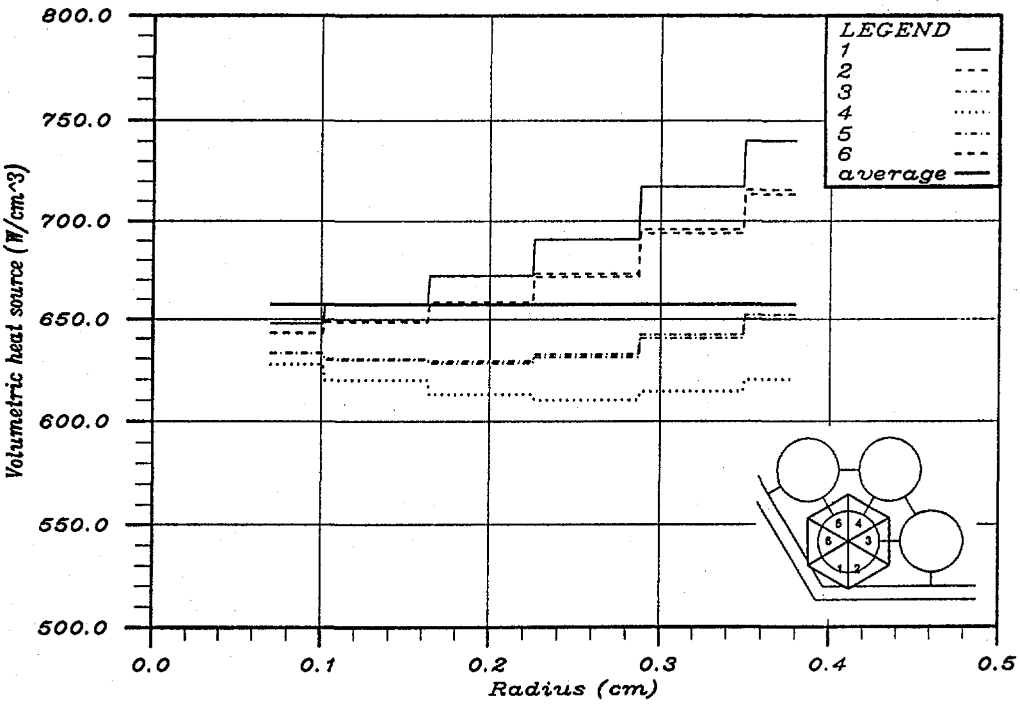
The results of the heat source and fuel temperature distribution analyses are presented in Figs. 3 and 4. The calculations prove the expectation that the fuel temperature is not sensitive and though the heat source alteration is relatively significant, it causes only small changes in the pellet temperature distribution. The most important result of the analysis is the determination of the typical change of the heat source distribution.

Table 1. Thermohydraulic parameters of the azimuthal subcells for one phase case

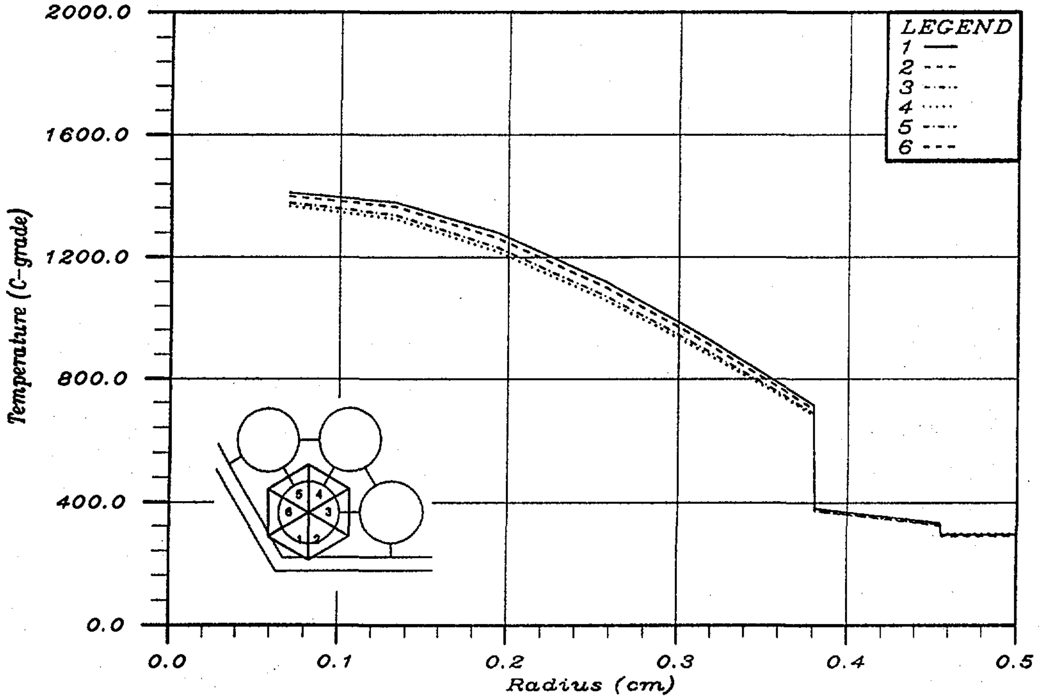
number of subcells	mass flux [kg/m^{2}s]	temperature [°C]	density [kg/m^3]	void fraction [%]
1	1959.	295.3	730.0	0
2	2566.	290.5	740.0	0
3	2555.	294.7	731.2	0
4	2544.	298.3	723.8	0
5	2554.	294.7	731.2	0
6	2567.	290.6	739.8	0

Table 2. Thermohydraulic parameters of the azimuthal subcells for two phase case

number of subcells	mass flux [kg/m^{2}s]	temperature [°C]	density [kg/m^3]	void fraction [%]
1	1891	326.9	456.4	33.3
2	2471	326.9	529.5	20.6
3	2305	326.9	463.5	32.1
4	2136	326.9	396.7	43.7
5	2310	326.9	463.5	32.1
6	2488	326.9	531.0	20.3

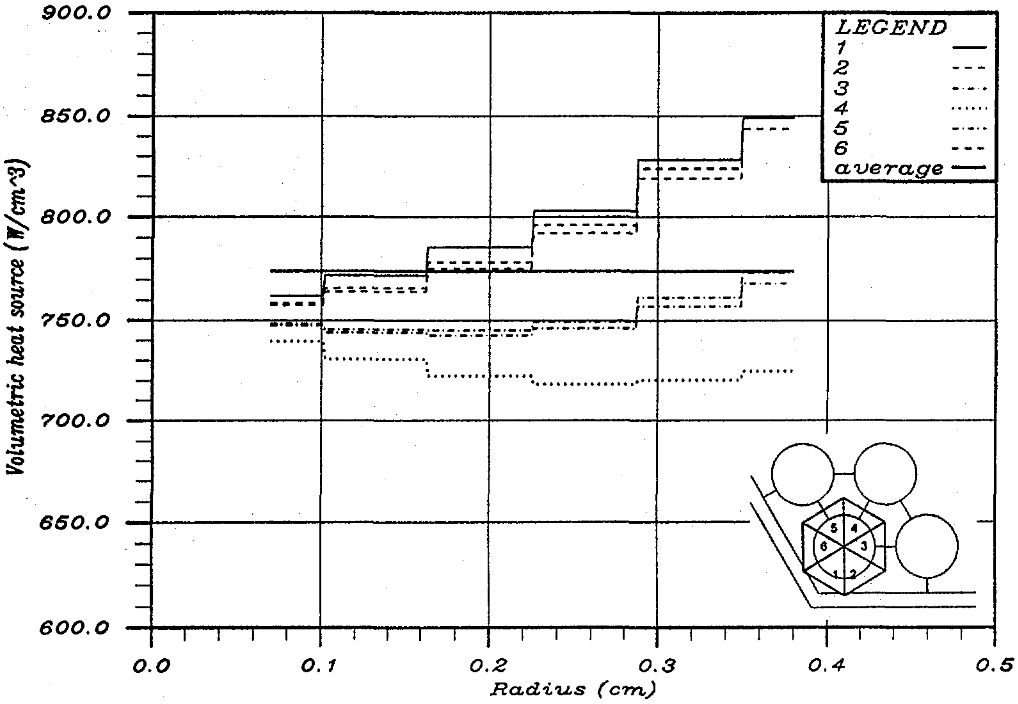


The volumetric heat source distribution in the pellet

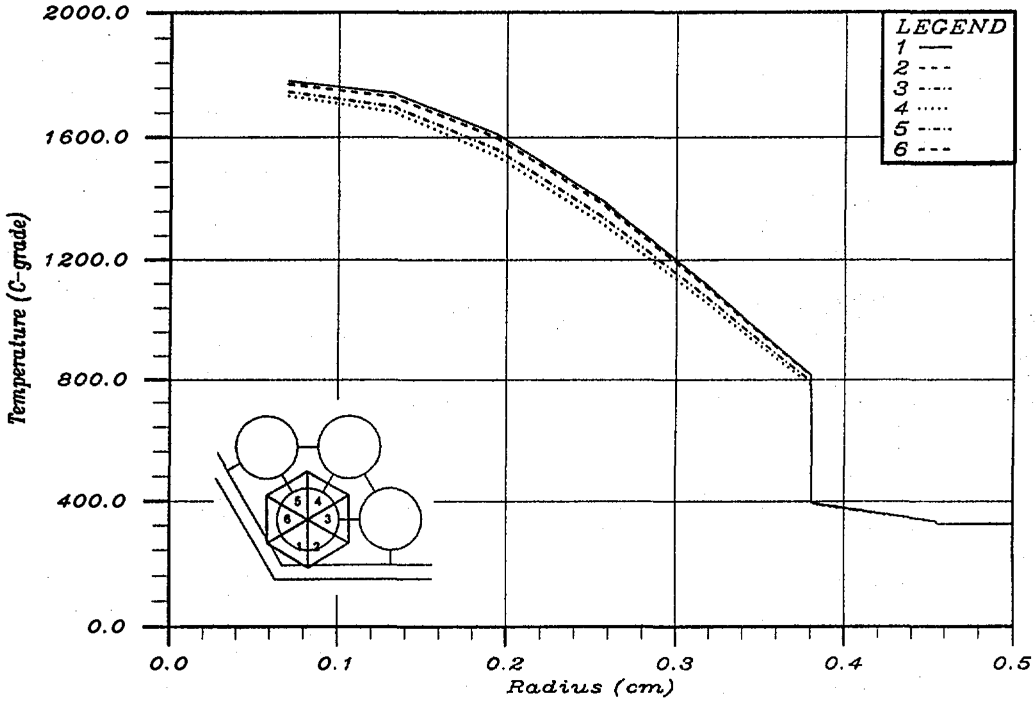


The temperature distribution in the fuel rod

Fig. 3. Single phase case (Case A)



The volumetric heat source distribution in the pellet



The temperature distribution in the fuel

Fig. 4. Two-phase case (Case B)

A interesting result of the analysis is that the coolant density alteration in the subchannels or subcells in the MGCP2D calculations has practically no influence on the distribution. It is not surprising, if it is taken in to account that the migration area of the neutrons is about 50 cm^2 while the area of the cell calculated by MGCP2D is 1-2 cm^2. The heat source distribution, in comparison with the average value of the power density, shows similar tendencies in both investigated cases and the deviations of the relative values are rather small.

5. Summary and conclusions

The effect of the non-uniform fuel pellet heat source distribution has been investigated. For the analysis a computer code system has been applied. The main program of the system was the KARATE-440 code, however, two new modules were added: a special 2D heat conduction code and a reactor physics module based on the calculation of collision probabilities. The heat source distribution in a fuel pellet can be determined by using the code system.

The results of the analysis revealed that there can be a strong non-uniformity of power distribution inside a fuel pellet, however, the effect depends first of all on the general assembly conditions, and the local subchannel parameters have only a slight influence on the pellet heat source distribution.

The results of the calculations indicated that, in spite of the fact that the power density distribution in a VVER-440 corner rod exhibits a significant inclination, the change of the surface heat flux of this fuel rod is less than four percent.

Finally, it has to be emphasized that the above result effects only one factor of the DNBR, i.e. it determines the local heat flux. The other factor of the DNBR, the value of the critical heat flux is also changing as a result of the alteration of the thermohydraulic parmeters in the subchannels.

6. References

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