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STUDY ON THE TRANSIENT BEHAVIOURS OF MNSR REACTOR FOR CONTROL ROD WITHDRAWAL

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摘　要

利用热工水力学程序 RETRAN-02 和反应堆物理计算程序 MARIA，计算和分析了微型中子源反应堆 MNSR 的瞬态特性。计算得到的事故序列和后果与实验值进行了比较。为了研究 Doppler 效应，考虑了反应堆的有效共振积分。计算了反应性温度系数的权重因子。计算了反应堆功率峰，冷却剂入口温度，出口温度和冷却剂质量流量等瞬态参数并与实验值进行了比较。

STUDY ON THE TRANSIENT BEHAVIOURS OF MNSR REACTOR FOR CONTROL ROD WITHDRAWAL

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ABSTRACT

The transient behaviours of Miniature Neutron Source Reactor MNSR are analyzed and calculated with the reactor thermohydraulics RETRAN-02 [1] program and the reactor physics MARIA [2] program. The obtained event sequence and consequence from the calculation are compared with the experiments [3]. The effective resonance integral for study on Doppler effect is taken into account. The reactivity temperature coefficient weighting factors are computed. The transient parameters related to reactor power peaking, coolant inlet temperatures, outlet temperatures and coolant mass flow, etc. are computed and compared with the experimental results.

1 DESCRIPTION OF MNSR REACTOR

1.1 Geometry and Composition

The fuel rod of MNSR reactor is composed of U-Al alloy pellet and Al cladding. The circles of fuel element are concentrically arranged in the light water coolant/moderator to set up the cylindrical core. The metal beryllium blocks are used for reflectors which have three types: (1) the side beryllium annulus surrounding the core, (2) the bottom beryllium plate, and (3) the top beryllium shims. The parameters of geometry and composition in the calculation are quoted from the reference [4].

1.2 Volumes, Junctions and Heat Conductions

All reactor in the vessel is divided into 8 volumes, 8 junctions and 3 heat conductions in the up-flow channel and the down-flow channel as shown in Fig. 1. In the up-flow channel the coolant passes through volume 1, junction 1, volume 2, junction 2, volume 3, junction 3, volume 4, junction 4, to volume 5, and in the down-flow channel, the coolant passes through volume 5, junction 5, volume 6, junction, 6, volume 7, junction 7, volume 8, junction 8 to volume 1. The coolant natural convection during the transient period is shown in Fig. 2.

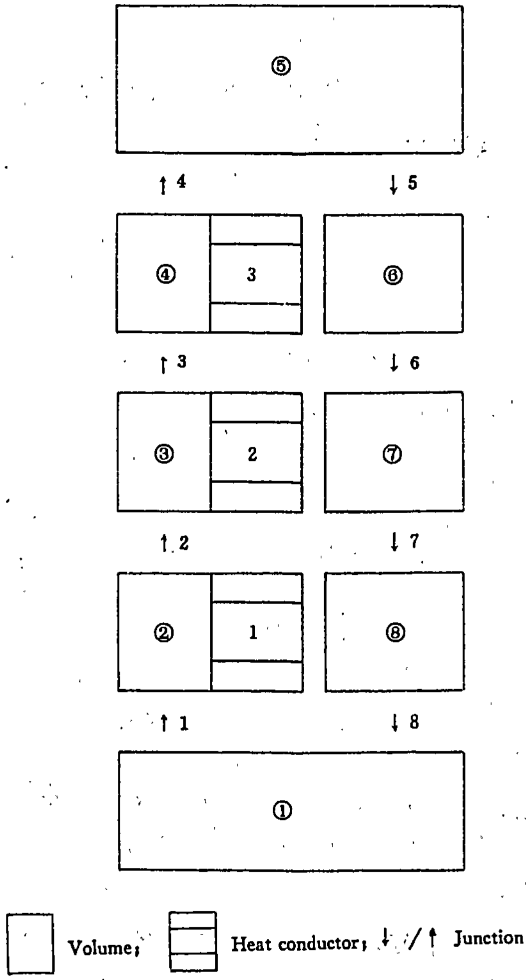

Fig. 1 Volumes schematic diagram

2 REACTOR INITIAL CONDITION AND TRANSIENT CAUSATION

Before the reactivity insertion, the reactor is in a subcritical state. The reactivity of reactor is controlled by the central control rod. The transient state is caused by the control rod withdrawal from the bottom to the top of core. The positive reactivity increases with the control rod withdrawal and results in the power and temperature ascension. The negative reactivity temperature coefficient of reactor suppresses and decreases the positive reactivity and returns the reactor to the subcritical state. The parameters relating to reactivity are listed in Table 1.

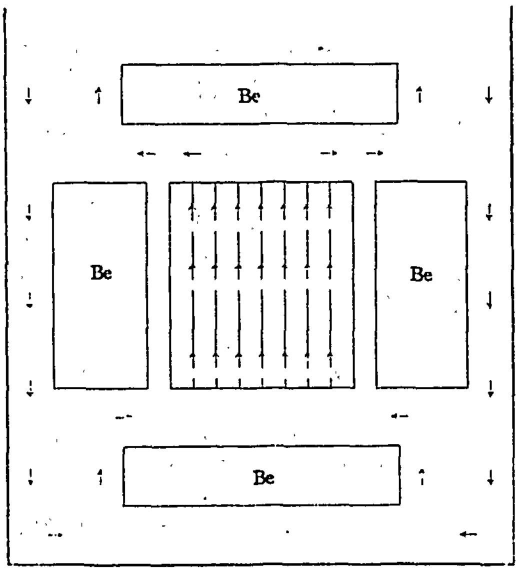

Fig. 2 Coolant natural convection in MNSR reactor

Table 1 Parameters relating to reactivity insertion

Efficiency of central control rod:	6.8 mk
Positive reactivity inserted:	3.6 mk
Effective delayed neutron fraction:	8.08×10^{-3}
Control rod withdrawal speed:	11.4 mm/s

3 EVALUATION FOR DOPPLER EFFECT AND WEIGHTING FACTOR

For evaluating Doppler effect, the effective resonance integral of MNSR fuel rod is computed by the following formula [5]:

$$I_{eff} = 4.35 + 11.9/(l \cdot \varepsilon) \tag{1}$$

where l is the average chord length of fuel rod, ε = relative nuclear number hole. The value of I_{eff} computed is nearly equal to resonance integral $280.0 \times 10^{-28}\mathrm{m}^2$ [6], so the resonance shielding effect and then Doppler effect are negligible.

Reactor average reactivity temperature coefficient is computed by the following formula [3]:

$$\alpha(t) = 0.026445 \cdot 10^{-3} - 0.0034752 \cdot 10^{-3} \cdot t \tag{2}$$

where t is the coolant/moderator average temperature.

The weighting factor of reactivity temperature coefficient is considered as:

$$W = \frac{\int_v \sum_g \Phi_g^+(r) \cdot \Phi_g(r) \cdot \mathrm{d}r/v}{\int_V \sum_g \Phi_g^+(r) \cdot \Phi_g(r) \cdot \mathrm{d}r/V} \tag{3}$$

where

Φ_g (r) =gth group neutron flux at space point r,

Φ_g^+ (r) =gth group function adjoint to the neutron flux at space point r,

g=neutron group number,

v=volume relative to control rod region,

V=core volume.

In the one-group neutron diffusion theory the weighting factor is written by:

$$W = \frac{V \cdot \int_{r_c} r \cdot J_0^2(Br) \cdot \mathrm{d}r}{v \cdot \int_R r \cdot J_0^2(Br) \cdot \mathrm{d}r}, \tag{4}$$

r_c=radius of control rod region,

R=radius of core,

J_0= (Br) =0th order Bessel function.

where B is the square root of reactor radial buckling which is calculated by MARIA program in Sn (n=4) method and 4 groups theory. The integrals in equation (4) are computed by Simpson rule and the results are given in Table 2.

Table 2 Reactivity temperature coefficient and Doppler weighting factor

Reactivity temperature coefficient/mk/℃	Doppler weighting factor/W
0.1126	1.389

The temperature distribution in coolant through the core is assumed to be linear, the inlet and outlet temperatures are obtained by the average temperatures extrapolation of the end volumes.

4 DESCRIPTION OF RETRAN-02 INPUT DATA FILE

Based on the transient behaviours of MNSR reactor to be calculated and the requirements of RETRAN-02 program for input data file, the 20 sets of input data card and the explanations are given in Table 3.

Table 3 Input data and explanations

Card No.	Variables	Explanations
01000X	X=1, 2, 3, 4, 5	Problem control and description
010005		Reactor power
01XX40	XX=01	Data tapes
02000Y	Y=1, 2	Minor edits
03XXX0	XXX=001, 002, 003	Time step data
04XXX0	XXX=001, 002	Trip control data
05XXXY	XXX=001, 002, …, 008 Y=1, 2	Volume data
06XXX1	XXX=001	Bubble data
08XXXY	XXX=001, 002, …, 008 Y=1, 2	Junction data
140000		Point kinetics constants
140XX0	XX=01, 02, 03	Reactivity coefficient
141XYY	X=0, YY=01, 02	Reactivity insertion data
1420XX	XX=01	Density reactivity table
1430XX	XX=01	Doppler reactivity table
1440XX	XX=00, 01, 02	Direct moderator heating data
15XXXY	XXX=001, 002, 003 Y=1, 2	Heating conductor data
16XXX0	XXX=001, 002, 003	Core section data
17XXYY	XX=01, YY=01, 02	Conductor geometry data
18XXYY	XX=01, 02, YY=00, 01, 02	Thermal conductivity data
19XXYY	XX=01, 02 YY=00, 01, 02	Volumetric heat capacity

5 RESULTS AND DISCUSSION

The transient bchaviours of reactor during the central control rod withdrawal from the core are computed by the programs and compared with the results measured on MNSR reactor. The reactor power peaking computed is a little greater than the experimental value about 1.5%. The coolant inlet/outlet temperatures are nearly the measurements. The comparisons between theory calculation and experiment are given in Table 4.

Table 4 Comparisons between theory calculation and experiment [3]

	Theory	Experiment
Power peaking/kW	76.9	75.8
Maximum mass flow/kg/s	0.45	0.49
Inlet temperature/℃	26.1	22.8
Outlet temperature/℃	63.0	62.8
Coolant average temperature/℃	42.2	
Difference between centerline and surface temperatures/℃	0.35	0.30

The event sequence in the transient period is listed in Table 5.

Table 5 Event sequence

Time/s	Event
0.0	Reactor in subcriticality
5.0	Control rod withdrawal, reactivity insertion
33.7	Maximum reactivity insertion
345.0	Reactor power peaking region (>70 kW)
465.0	Volume maximum temperature
695.0	Maximum element surface temperature
770.0	Maximum mass flow
1135.0	End of problem

It can be seen from the results calculated and comparisons with the experiments, that the power and temperatures in the transient period are a little over evaluation, the consequences are obtained by the programs conservatively. Even in the case of severe accident caused by the control rod all over withdrawing from the MNSR core, it can automatically return the supercriticality to the subcriticality by itself, the inherent safety of MNSR reactor is further verified.

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