CÁLCULO NEUTRÓNICO DE PARÁMETROS DE SEGURIDAD DE LOS REACTORES RP-10 Y RP-0

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SUMMARY

Theoretical safety calculations were done with proved codes utilized by the staff of the RERTR program in the HEU to LEU core conversions. The studies were designed to evaluate the reactivity coefficients and kinetics parameters of the reactor involved in the evolution of peak power transients by reactivity insertion accidents. It was done to show the trend of these reactivity coefficients as a function of the core size and fuel depletion for RP10 cores. It was useful to get a better understanding of the progression of the reactivity insertion transients monitoring the critical thermal-hydraulic parameters to avoid core damage. To confirm the accuracy of these studies the results were compared with experimental data of the SPERT I reactors. The microscopic cross section calculations were condensed to 15 broad groups using the WIMSD4M[1] code for all the isotopes from eight different regions that model seven different assemblies. The Supercell, multiplate and homogenized options were used to represent the different assemblies in the reactor. For diffusion theory calculations the DIF3D[2] code was used in planar geometry with input axial buckling to simulate axial leakage. To Benchmark the designed models used in the cross sections generation and the DIF3D designed model, the VIM[3] Monte Carlo code was used. The RECOEFF[4] code was used to calculate the reactivity coefficients.

1. INTRODUCTION

The neutronic parameters calculated were power density, the beta effective, fast neutron life time & prompt neutron generation time. The reactivity coefficients of moderator temperature, fuel temperature and moderator density have also been calculated. These parameters, which have relation with safety analysis, have been studied to show the trend as function of the core size and core burn-up..

2. CALCULATIONAL MODELS

For the neutronic analysis of the RP10 reactor, the RP10 A16/25 and RP0-16/12 core configuration have been represented by seven basic assemblies. Each of these assemblies has been modeled using the WIMS-D4M code to get the microscopic cross sections of all the most important isotopes of each material in each assembly. To get these results, the new WIMS-D4M Supercell option and the multiplate and homogenized options have been applied.

The Supercell option was used for the control rod guides and extra region of the control fuel element, for the graphite and beryllium reflectors, irradiation box, water box, and fine control rod box. For the standard fuel element calculations the simple cell option was used. The Supercell and Multiplate options were used for the control rod guides and extra region of the control fuel element. All the WIMS-D4M output files that have the microscopic cross sections for all the major isotopes were merged to one binary file used the input for DIF3D code diffusion calculations. These calculations have been made with depleted and fresh fuels assuming that all the control rods were fully withdrawn from the core. The water temperature coefficient is calculated for temperatures ranged from 300K to 500K and temperatures for fuel from 300K to 800K.

Generation of the Microscopic cross sections

For all the basic assemblies, the WIMSD4M integral transport calculation was performed in a structure of 30 groups in which 16 are thermal groups. These were condensed to 15 energy groups for the core calculations. The density and composition for each assembly of the RP10 & RP0 reactors are described in the next tables 1 to 5.

Table 1. Fine control rod box.

Element	Density(g/cm^3)	Area(cm^2)	N(atoms/b.cm)
Al	2.700000	18.664000	1.7813439e-2
C	1.710000	38.080000	5.1708259e-2
H2O	0.997815	6.396000	6.758251e-3 (H)
			3.378786e-3 (O)
Total	**1.930498**	**63.140000**	

Table 2. Irradiation box.

Element	Density(g/cm^3)	Area(cm^2)	N(atoms/b.cm)
Al	2.700000	20.030200	1.9117379e-2
H2O	0.997815	43.109800	4.5551e-2 (H)
			2.2773e-2 (O)
Total	**1.537807**	**63.140000**	

Table 3. Graphite Reflector box.

Element	Density(g/cm^3)	Area(cm^2)	N(atoms/b.cm)
Al	2.700000	9.436400	9.00636e-3
C	1.710000	51.063500	6.93361e-2
H2O	0.997815	2.640100	2.78968e-3 (H)
			1.39467e-3 (O)
Total	**1.930498**	**63.140000**	

Table 4. Beryllium Reflector box.

Element	Density(g/cm^3)	Area(cm^2)	N(atoms/b.cm)
Al	2.700000	7.311600	6.89828e-3
Be	1.850000	52.375700	1.025457e-1
H2O	0.997815	3.452700	3.64825e-3 (H)
			1.82394e-3 (O)
Total	**1.898240**	**63.140000**	

Table 5. Fuel Assembly.

Element/MEAT	Density(g/cm^3)	N(atoms/b.cm)
U3O8	2.714850	
U(235+238)	2.317246	
U235	19.75%	1.172575e-3
U238	80.25%	4.704334e-3
Al-27	37.7%(4.3577)	3.666676e-2
O-16	9.6%(4.3577)	1.574619e-2
Element/**CLAD**	Density(g/cm^3)	N(atoms/b.cm)
Al-27	2.7000	6.02630e-2
Element/**WATER**	Density(g/cm^3)	N(atoms/b.cm)
H	11.1904%(1.0)	6.6859193e-2
O-16	88.8096%(1.0)	3.3427938e-2

WIMSD4M Models for Microscopic Cross Sections Calculations

The 7 designed models were used in the WIMSD4M[1] code. The input for SFE and CFE cross sections generation was prepared according an automation option[10]. For diffusion calculations the reactors were modeled in three dimensions using the DIF3D code. The reactivity coefficients were calculated from core Keff data for all the cases at the different water densities and water temperatures (from 300K to 500K) and fuel temperatures (from 300K to 800K). The WIMSD4M/DIF3D input files were prepared with the purpose of Automation of reactivity Coefficient Calculations[10]. To Benchmark the models used in the cross sections generation and the DIF3D diffusion theory model the VIM[3] Monte Carlo code was used. The results presented in the table N^{o} 6 show good agreement although these calculations were performed with cero axial leakage.

Table 6.

	RP10 reactor	**RP0 reactor**
	K_{eff} Φaxial=0	K_{eff} Φaxial=0
DIF3D	1.206550	1.102036
VIM	1.217200	1.109400

SAFETY PARAMETERS

Reactivity coefficients as a function of moderator temperature, moderator density and fuel temperature has been calculated for RP10 and RP0 reactors at the beginning for

fresh cores and burned core for RP10-A16/25(core 16). Two dimensional core calculations were performed using the WIMSD4M and DIF3D codes performed in 15 energy groups although the use of fewer broad group structure provides[8] (less negative) reactivity coefficients. The reactivity coefficients for the two cores are showed in the graphics 1, 2. The reactivity coefficients of moderator temperature, moderator density and fuel temperature show to be more negative for burned cores (Table 7). This evolution can be appreciated for the RP10 core. The RP0 reactor calculations were done for BOC only. For RP10 and RP0 fresh core calculations, the moderator density and the fuel temperature reactivity coefficient show to be more negative for RP0 core. The reason of this effect is due to the increased leakage as a result of the harder spectrum and shift of power profile. The Doppler reactivity coefficients show to be more negative for the RP0 fresh core although both RP0 and RP10 cores have the same quantity of U^{238}. In these LEU fueled cores, the feedback response is almost exclusively due to the coolant and fuel temperatures and coolant density effects. In the accidents analysis the two former shows to be delayed effects, while the fuel temperature is instantaneous having an important reactivity control effect at the beginning of the peak power transient evolution.

Table 7. DIF3D calculations.

	RP-10	A16/25	RP0 A16/12
Coefficient	Fresh	Burned	Fresh
Coolent Temp. *	-0.00990	-0.01400	-10.16e-03
Coolent Density **	-0.18200	-0.19600	-0.32000
Sum Coolent Coeff.	-0.21680	-0.22050	-0.36020
Fuel Temperature *	-0.00168	-0.00185	-0.00217
Total Coefficient	-0.21848	-0.22224	-0.36225

* Units in $\%\Delta K/K/\Delta T^0C$, ** Units in $\%\Delta K/K/\%$void. The coolant temperature coefficient calculated at 40^0C. The coolant density coefficient is calculated changing the density 1%. The fuel temp. coefficient is calculated at 100 °C.

The WIMSD4M/DIF3D results show a close agreement with experimental data[4].

Table 8. Comparison of the RP10 calculated values with experimental data.

	$\%\Delta K/K/\%$void	$\%\Delta K/K/\Delta T^0C$ Moderator	$\%\Delta K/K/\Delta T^0C$ Fuel
DIF3D	1.96000e-1	8.60e-3	1.850e-3
Experimental data	2.0076e-1	4.2666e-3	1.7440e-3**

*The moderator temperature coefficients are calculated with a moderator temperature of 25 ^{0}C. The voiding coefficient was calculated for 1% density change. **Theoretical value[5].

The Table 9 has the Beta effective of the delayed neutrons, neutron prompt generation time and neutron prompt lifetime, Table N$^{\underline{0}}$ 10. These values were evaluated with the VARI3D [9] code.

Table 9. Kinetics parameters calculated with VARI3D code.

		β_{eff}	Generation Time(s)	Prompt lifetime
RP10	Fresh core	7.725434e-3	5.20033e-5	5.85768e-5
	Burned core	7.544569e-3	5.88546e-5	6.07199e-5
RP0	Fresh core	8.193778e-3	6.14437e-5	6.23870e-5

SAFETY CALCULATIONS

The safety studies have been performed with the PARET code[7]. The values of the table 10 were used in these evaluations.

Table 10. Safety calculations with PARET code. The case (a) is run with the kinetics parameters values considered in the ref.[5]. The case (b) is with the values of the table No 12.

$\rho_{inserted}$($) 1.5/ 800ms	τ(1/s)	Peak Power (MW)	$T_{CLA}(^0C)$ peak Power.	$T_{FUE}(^0C)$ peak Power	Time(s) peak Power
RP10 \ a)	67.90	369.10	227.53	308.70	0.894
\ (b)	64.85	322.35	212.01	286.31	0.905
RP0	67.22	130.46	177.94	243.83	0.8796

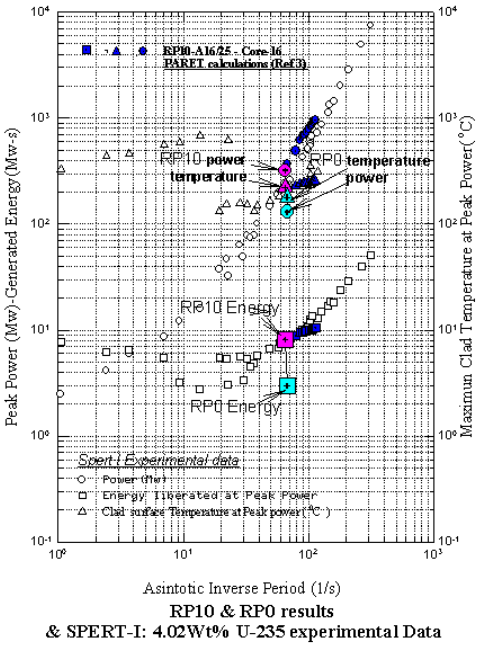
The table 10 contains the values of PARET calculations at the time of peak power. For validation purpose the results were compared with experimental data of the SPERT I reactor.

These values are plotted and showed in the fig N$^{\underline{0}}$ 3. There is a good agreement of the RP10 & RP0 reactors for insertions accidents

compared with the experimental data of the SPERT-I Low enrichment core[11].

3. CONCLUSIONS

These validated neutronic calculations have established credible models and procedures to analyze the trend of the reactivity coefficients and safety calculations for the RP10 and RP0 reactor. For the same configuration the trend of the reactivity coefficient shows to be more negative for burned cores than fresh cores. So the worst case for the safety analysis study of a burned core can be done with the kinetics parameters of fresh core that show to be rather less negative. At the same way the peaking factor is grater for fresh than burned cores. A small core like the RP0 shows to have better reactivity coefficients (more negative) than a greater core like the RP10. This report shows that the analyzed transients don't cause any damage.



RP10 & RP0 results
& SPERT-I: 4.02Wt% U-235 experimental Data

Fig N° 2

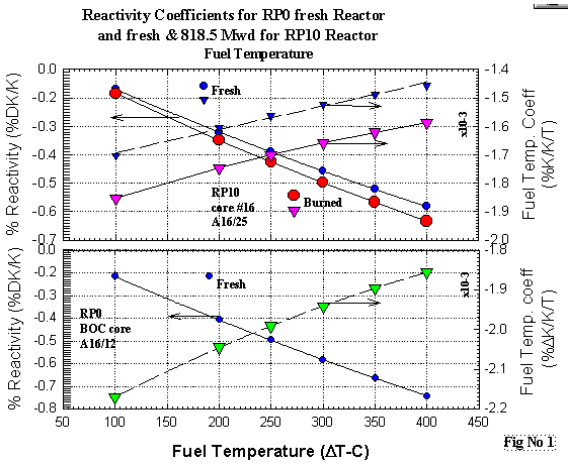


Fig No 1

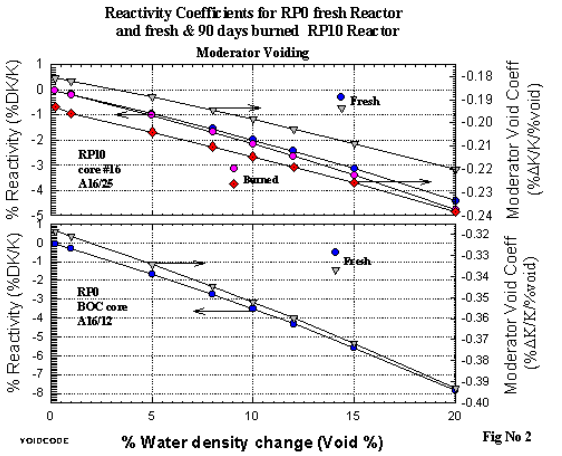


Fig No 2

4. References

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