CORE CALCULATIONS FOR THE UPGRADING OF THE IEA-R1 RESEARCH REACTOR

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Presented at the

1998 International Meeting on

Reduced Enrichment for Research and Test Reactors

October 18-23, 1998

São Paulo, Brazil

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ABSTRACT

The IEA-R1 Research Reactor is a multipurpose reactor. It has been used for basic and applied research in the nuclear area, training and radioisotopes production since 1957. In 1995, the Instituto de Pesquisas Energeticas e Nucleares (IPEN/CNEN-SP) took the decision to modernize and upgrade the power from 2 to 5 MW and increase the operational cycle. This work presents the design requirements and the calculations effectuated to reach this goal.

INTRODUCTION

In 1995, in view of a favorable budget and the priorities given to the production of some useful radioisotopes, the Instituto de Pesquisas Energeticas e Nucleares (IPEN/CNEN-SP) took the decision to modernize and upgrade the power of the IEA-R1 reactor from 2 to 5 MW and increase its operational cycle from 8 h/day, 5 days a week to 120 h continuous per week. The IEA-R1 Research Reactor is a multipurpose reactor. It has been used for basic and applied research in nuclear area, training and also for radioisotopes production. At the new power level and operational strategy the IEA-R1 research reactor will become actually a radioisotope producer satisfying part of the demand for primary radioisotopes in Brazil. As a consequence of the increase of the level of neutron flux, other applications such NTD silicon production, neutron radiography and neutron activation analysis will have better performance.

To accomplish safety requirements, a set of actions was performed following the recommendations of the IAEA Safety Series 35 [1] applied to research reactors. Such actions consisted in the modernization of old systems, design of new ones, safety evaluations and licensing and elaboration of experimental/operational routines to be submitted and approved by the Safety Review Committee.

FUEL ELEMENT DESIGN

The conversion of IEA-R1 Reactor from HEU to LEU has started in late 1988 with the introduction of the first Brazilian made fuel element of U_3O_8-Al dispersion type with 1.9 gU/cm^3 [2]. The strategy was to substitute gradually the HEU fuel to LEU fuel. Having a heterogeneous core (HEU and LEU), the design decision was made to have identical geometry (plate thickness, width and pitch between plates) for both fuel assemblies, and to have the same quantity of ^{235}U in the fuel plates (10 g /fuel plate; 180 g /fuel assembly). The conversion from HEU to LEU proceed up to 1997 when the reactor had to be fitted for the upgrade from 2 to 5 MW.

In order to optimize the neutron flux and to have enough reactivity for continuous operation profiles strategy , the size of the core was changed from 30 to 25 fuel assemblies, and the uranium content of the fuel plate increased to 2.3 gU/cm^3. The design decision was to increase uranium content and to maintain the fuel plate geometry, this means to increase the uranium compound fraction in the dispersion. The core was completely converted from HEU to LEU but still having a heterogeneous core with different uranium fuels plate contents.

In order to increase each fuel reactivity and to lower the number of fuel elements in core and the number of fuel assemblies to be changed (and fabricated) during the year, dispersion fuel of U_3Si_2-Al with 3.0 gU/cm^3 will be also used in the reactor. Again, the design decision is to keep the fuel plate and fuel assembly geometry identical as the U_3O_8-Al fuels. The replacement of fuel assemblies will be done in a continuous way, having a heterogeneous core up to the equilibrium core.

In increasing the reactor power from 2 to 5 MW thermal /mechanical /fuel performance corrosion / etc. analysis had to be done. For each reactor operational condition, the fuel assembly functional requirements were demonstrated to be attended and the parameters of interest within limits and margins established.

One relevant point is the cladding corrosion rate. IPEN fuel uses Al-1060 as cladding for the fuel plate. For 5 MW the cladding temperature is higher than 2 MW and the corrosion rate will be also higher. It's planned to increase resistance to corrosion by changing the cladding from Al-1060 to Al-6262. Table I presents the actual fuel specifications.

Table I - General Fuel Element Assembly Specification

	U_3O_8 Powder	**U_3Si_2 Powder**	
Particle Size	< 89 µm ; maximum of 20% < 44 µm	< 89 µm ; maximum of 20% < 44 µm	
Particle Density	> 8.0 g/cm^3	> 11.7 g/cm^3	
Specific Surface	<0.1 m^2/g	<0.15 m^2/g	
Enrichment	19.75 ± 0.2 wt%	19.75 ± 0.2 wt%	
	U content > 84.5%	Si content 7.5+0.4/-0.1 %;U_3Si_2 content >80%	
Maximum Impurities Level (µg/g)	Al-250;B-2;C-500;Ca+Mg-200;Cd-0.5;Cl+F-350;Cr-200,Co-3;Cu-250;Fe-250;Li-5;Mn-250;Mo-250;N-200;Ni-200;P-250;Pb-250;Si-250;Sn-250;Th-10;Ti-250;V-250;Zn-250,W-250	Al-600; B-10; C-200; Cd-10; Co-10; Cu-500; Fe+Ni-1500; H-200; Li-10; N-2500; O-7000; Zn-1000; others -500	
Maximum humidity	< 1%		
Al Powder			
Particle Size	< 44 µm		
Al Content	> 99%		
Al_2O_3 Content	< 0.7%		
Impurities	< 0.1%		
Chemical Composition Limits (ppm)	Cu-20000;Fe+Si-95000;Mn-5000;Zn-10000;Other-5000;B-10;Cd-10;Li-10;Co-10		
Compact			
	U_3O_8-Al (1.9 gU/cm^3)	**U_3O_8-Al (2.3 gU/cm^3)**	**U_3Si_2-Al (3.0 gU/cm^3)**
U Compound Mass	58.5 ± 0.2 g	69.7 ± 0.3 g	85.8 ± 03 g
Al Mass	46.8 ± 0.2 g	41.3 ± 0.2 g	46.8 ± 0.2 g
Density	4.1 ± 0.2 g/cm^3	4.3 ± 02 g	5.2 ± 02
Nominal Dimensions (mm)	104.2 x 59.1 x 4.2		
Picture Frame and Cladding			
Material	Aluminum , alloy 1060		
Fuel Plate			
Homogeneity			
Nominal	26.8 mg^{235}U/cm^2	32.4 mg^{235}U/cm^2	42.6 mg^{235}U/cm^2
Maximum Central Deviation	± 12%		
Maximum Extremities Deviation	± 25%		
Blister Test	500 °C/1 hour without any blister		
Dimensions (mm)			
Internal Plate Dimensions	625 x 70.75 x 1.52		
External Plate Dimensions	714 x 70.75 x 1.52		
Active Region Dimensions	590 (min.) x 60.35 (min.) x 0.76		
Cladding Thickness	0.38 (nominal) ; 0.25 (min.)		
Defects			
White Points	< 0.5 mm of diameter. ; > 0.4mm from plate border		
Cladding Surface	< 0.1 of depth		
Surface Contamination	< 10 µg of U per plate		
Side Plates, Bottom Nozzle and Handle Pin			
Material	Aluminum – alloy 6262 T6		
Fuel Element Assembly			
Fuel Plate Extraction Force (mechanically assembled)	20 to 30 N/mm		
Water Channel Width	2.89 mm		

NEUTRONIC CALCULATIONS

Figure 1 shows the standard configuration of the IEA-R1 reactor core with 21 standard fuel elements and 4 control elements and Figure 2 shows the ^{235}U burnup distribution for the design of the first core at 5 MW (September 1997). The standard fuel elements (FE) numbered 153 to 159 contain 2.3 g U/cm^3 while the remainders contain 1.9 g U/cm^3.

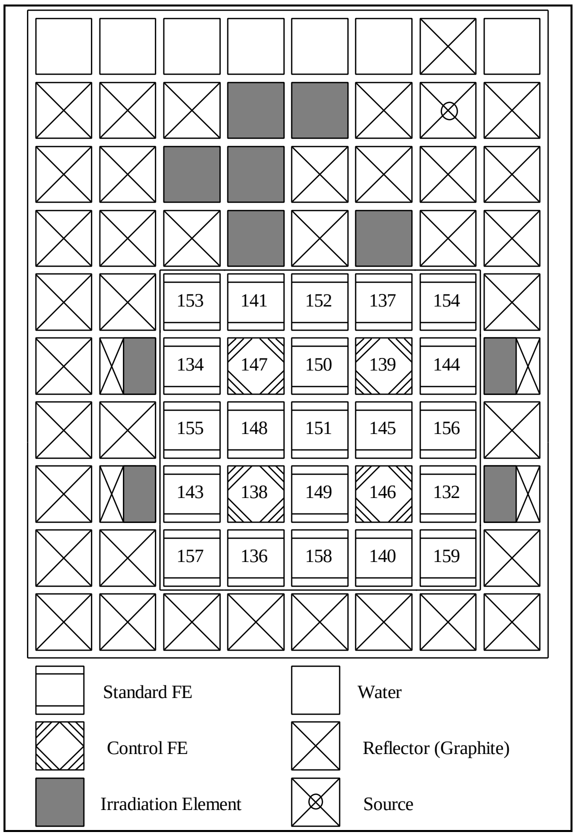

Figure 1. Standard Configuration.

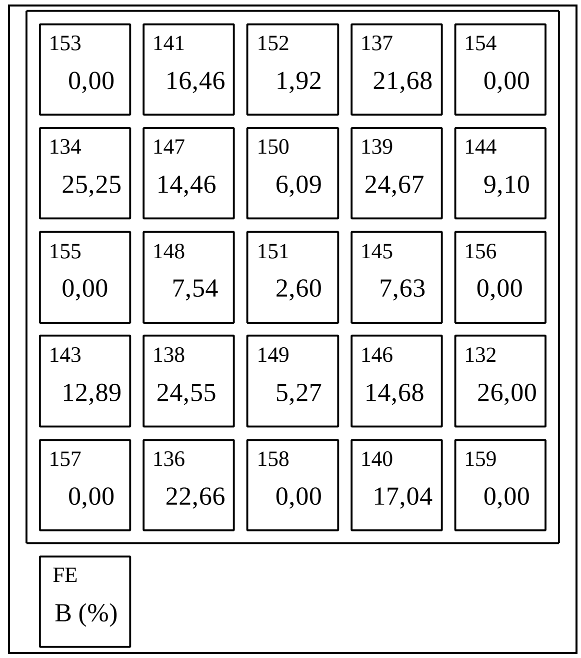

Figure 2. ^{235}U Burn-up Distribution.

The calculational methodology is based on LEOPARD [3, 4] and HAMMER-TECHNION [5] codes for cross-section generation, 2DB [6] code for the core and burn-up calculations in a two-dimensional geometry and CITATION [7] code for a three-dimensional analysis. The fuel cross-section is performed with LEOPARD (version modified by Michigan University, where a plate geometry option was included) using a standard cell model (fuel, cladding and moderator) with an extra region to take into account other regions of the fuel element. The HAMMER-TECHNION is used to generate the cross-sections for the non-fuel regions such as reflector, control rods, etc. The reactor power history is simulated with 2DB in a two-dimensional model. Three-dimensional calculations is finally made with CITATION for effective multiplication factor, neutron flux and power density distributions, integral and differential control rod worth, reactivity coefficients and kinetic parameters.

The core reactivity excess requirements for 6 cycles of 5 days of continuous operation are:

- xenon equilibrium 3200 pcm;
- restart up to 1 hour after scram 800 " ;
- power defect 300 " ;
- operation control margin 500 " ;
- 6 cycles of 5 days of continuous operation 1800 " ;
- experiments 1000 " ; and
- total 7600 pcm.
-

The purpose of the design criteria is to assure that the reactor operation is safe and controlled so that the reactor can be shut down and held subcritical for all operational states and that safety limits are not exceeded. Thus, the following design criteria have been established based on the Safety Series-35 [1]:

- at least 200 % of the maximum reactivity excess shall be available in the reactivity control mechanisms. This criterion assures a shutdown margin of 100 % over the reactivity excess;
- the reactor shall be maintained subcritical when the most reactive control rod is fully-withdrawn and the others are fully-inserted (stuck-rod criterion). In this condition the effective multiplication factor shall be less than 0.98;
- the maximum rate of addition of positive reactivity shall be less than 35 pcm/s; and
- the temperature and void reactivity coefficients shall be negative.

Table II shows the main neutronic parameters for the core configuration for 5 MW. The effective multiplication factor satisfies the excess reactivity requirements to 6 cycles of 5 days of continuous operation. The control rod worth is at least twice the reactivity excess, therefore, assures a shutdown margin greater than 100 %. The maximum rate of addition of positive reactivity by control rod withdrawn is less than the value specified on design criterion. All reactivity coefficients are negative within the operation temperature range. Table II shows also the kinetic parameters calculated by CITATION code in a standard 4 energy groups structure.

Table II. Neutronic Parameters

Parameter	Value
Effective multiplication factor (k_{eff}), all control rods out	1.07594
k_{eff}, all control rods in	0.89643
k_{eff}, all control rods in without most reactive	0.95201
control rod worth	18784 pcm
maximum reactivity insertion rate by control rod withdrawn	26.4 pcm/s
average fuel temperature coefficient (α_F) from 20°C to 100°C	-1.91 pcm/°C
average moderator temperature coefficient (α_M) (20°C - 80°C)	-12.26 pcm/°C
average moderator density coefficient (α_{MD}) (20°C - 80°C)	-10.42 pcm/°C
average void coefficient (α_V) (0 to 2.7 % void)	-231.92 pcm/% void
effective delayed neutron fraction (β_{eff})	0.00763
prompt neutron generation time (Λ)	57.90 μs

THERMAL HYDRAULIC CALCULATIONS

The purpose of thermal-hydraulic calculations is to estimate temperatures, heat fluxes and flow rates at the fuel elements of the core and to verify if these parameters are within the design limits.

The calculational methodology is based on COBRA 3C/RERTR [8] code. This code is a modified version of COBRA 3C/MIT. The fuel heat transfer model considers radial conduction within the fuel and convection to the cooling fluid. Axial conduction at the meat and at the cladding is neglected. The fuel is divided into 25 equally spaced nodes in the axial direction.and 6 nodes in the radial direction (5 for the meat and one for the cladding). The coolant fluid is also divided in 25 nodes in the axial direction.

The design criteria must assure that, during operational state, adequate core cooling capacity will be available to keep the reactor fuel with adequate thermal safe margins. The basics information on thermal-hydraulic core design, based on the correlations of the TEC-DOC 233 [9] and the recommendations of the Safety Series-35 [1], include the following:

- The coolant temperature shall be less than the saturation temperature;
- The average temperature at the clad surface shall be less than 95^0 C, to limit the rate of corrosion;
- The peak clad surface temperature shall be less than the ONB temperature;
- The coolant velocity shall be limited to 2/3 of the critical velocity;
- The peak heat flux shall be less than the heat flux for flow instability;
- The peak heat flux shall be less than the burnout heat flux.

The heat flux for flow instability was calculated by the correlation proposed by Whittle and Forgan [10]. The maximum coolant velocity, for design purposes is recommended to be limited to 2/3 of the critical velocity which was estimated by Miller[11]. The clad surface temperature over which nucleate boiling may occur was calculated by the correlation developed by Bergles and Rohsenow [12]. To determine the minimum DNBR the Labuntsov [13] correlation was used.

The simulations performed were based on conservative assumptions where all uncertainties are considered at the same time. Table III shows the uncertainties considered which are related to the temperature and flow rate measurements, geometric tolerances and model uncertainties.

Table III – Uncertainties Considered

Parameters	**Uncertainties**
Flow Rate	10%
Inlet Temperature	2^0C
Power measurement	5%
Meat thickness	10%
Repartition uranium	12%
Uranium content	2%
Thickness channel	10%
Neutronic model	10%
Over power	10%

Table IV shows the main safety margins obtained for this core considering all the uncertainties.

Table IV – Main Safety Margins

Parameters	Calculated Value	Limit Value	Safety Margin Percentual
Coolant Temperature	62^0 C	112^0 C (saturation)	80%
Peak Surface Temperature	94^0 C	118^0 C (ONB Temperature)	24%
Average Surface Temperature	72^0 C	95^0 C	32%
Heat Flux for Flow Instability	32.7 W/cm^2	105.6 W/cm^2	223%
Flow Velocity	2.11 m/s	15 m/s	610%
MDNBR	9.0	2.0	350%

Figure 4 shows the temperature distribution along the hot channel, the coolant temperature at the channel outlet and the ONB temperature. The maximum temperature, obtained at the maximum flux region (0.80 m from the top of the plate), is 96^0 C for the meat and 94^0 C for clad surface. These temperatures are far below the ONB temperature, which is 120^0 C at the same position.

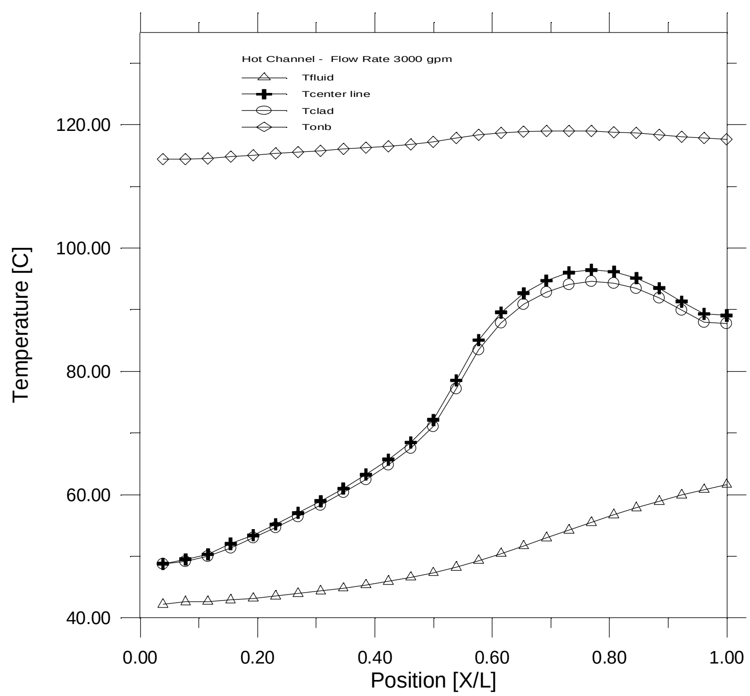

Figure 4 -Temperature Distribution

CONCLUSION

In the upgrading of the IEA-R1 Research Reactor, the core was completely converted from HEU to LEU. Its size was changed from 30 to 25 fuel elements in order to optimize the neutron flux. Also, the uranium content of the fuel plate was increased to 2.3 gU/cm^3. Neutronic, thermalhydraulic and fuel performance analyses of the IEA-R1 core for 5 MW showed that all criteria are within the limits and margins established.

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