PRELIMINARY ANALYSIS OF CONTROL ROD ACCIDENTS IN THE CRCN-R1 MULTIPURPOSE REACTOR CORE OF RECIFE IN BRAZIL

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The paper shows some results of the neutronic accident analyses arised by uncontrolled control rod withdrawal, based on the Conceptual Project of the CRCN-R1 MultiPurpose Reactor of Recife. In that reactor, a project of the CNEN/Brazil, under the leadership of the IPEN/São Paulo, is verified the thermal hydraulic limits in the reactor core during transients that simulate startup and power operation accidents. It has utilized a computer program that solved the kinetic equations based on multigroup diffusion theory, in our case we have used 4 energy groups, Two-Dimensional X-Y in the space, and 6 groups of delayed neutrons. A simple model of feedback is admitted in the capture and scattering macroscopic cross sections, in the fuel regions, temperature and coolant densities dependents. Based on those models, the results demonstrated that the reactor exhibits good degree of safety.

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INTRODUCTION

The multipurpose reactor porposed for the CRCN (Regional Center of Nuclear Sciences), is planned to be a pool type reactor with 20 MWth, cooled by light water and, in principle, reflected by Beryllium elements. Its conception is shown in the references /1,2,3/. For the fact of the reactor to be constituted of two subcritical arrangements, a preliminary analysis for some transients is necessary, to prevent possible unbalance of its critical state, and as consequence, their reflexes in the behavior thermal hydraulic of the core. As in the phase of the conception of the core there is a very intimate iteration between the neutronic and the thermal hydraulic, our analysis is centered in the most decisive transients for an alteration of design, in other words, the short duration transients, such as the startup and power operation accidents, caused by malfunction of control rod banks. In the calculations is used a version of the DINUCLE code, a program for transient calculations in reactor cores, based on multigroup diffusion theory, for problems in cylindrical or Cartesian geometries, even bi-dimensional, with several energy groups for the prompt neutrons and precursor groups of delayed neutrons /4 /. The thermal hydraulic equations are resolved simultaneously with the kinetic equations, in the coolant channels of the fuel plates in the reactor core. In that way, we can consider the feedback effects with the temperatures in the macroscopic cros sections, at the fuel region of the core. In special, in the capture and scattering cros sections, in the attempt of observing the Doppler effect in the fuel and the moderator effect in the coolant, as described in the reference /5 /.

REACTOR CORE

The reactor core is cooled by light water, moderated and reflected by a combination of D_2O tank and beryllium elements. Its important characteristic is to have a core divided into two halves coupled by a central heavy water tank. This tank provides a large region with high thermal neutron flux and furnishes an additional safety feature, which is to shut the reactor down when it is quickly emptied. This additional shutdown capability has a different engineering principle from that of standard safety and control rods and, therefore, enhances the overall reactor safety.

The D_2O tank has a region with thermal neutron flux magnitude of 10^{14} n/cm^{2}s, which is very appropriate for producing radioisotopes, obtainig neutron spectra for boron neutron capture therapy, obtaining neutron spectra for cold neutron experiments and many other applications. The beryllium refletor improves the fuel utilization and yields neutron fluxes adequate for materials and fuel irradiation.

The reactor core has 30 plate type (MTR) fuel elements, divided into two halves coupled with, in this work, a 30 cm heavy water tank, surrounded by beryllium reflector elements. The table 1 shows the main core characteristics, in agreement with the representation of the reactor core, in the figure 1, following.

Table 1

DIMENSIONS	cm
Core width – Direction x	99.7
Core length – Direction y	143.4
Reflector layer thickness	2x10.0
Zircaloy thickness	1.5
Beryllium layer thickness	2x8.1
Dimensions of the fuel regions	15x2.7 by 3x2.7
Dimensions of the control materials	6x2.7 by 1.4
Dimensions of the aluminium guides	3x2.7 by 1.4
Minimum width of the heavy water tank	30.0
Thickness of A,B and C in the control regions	1.4

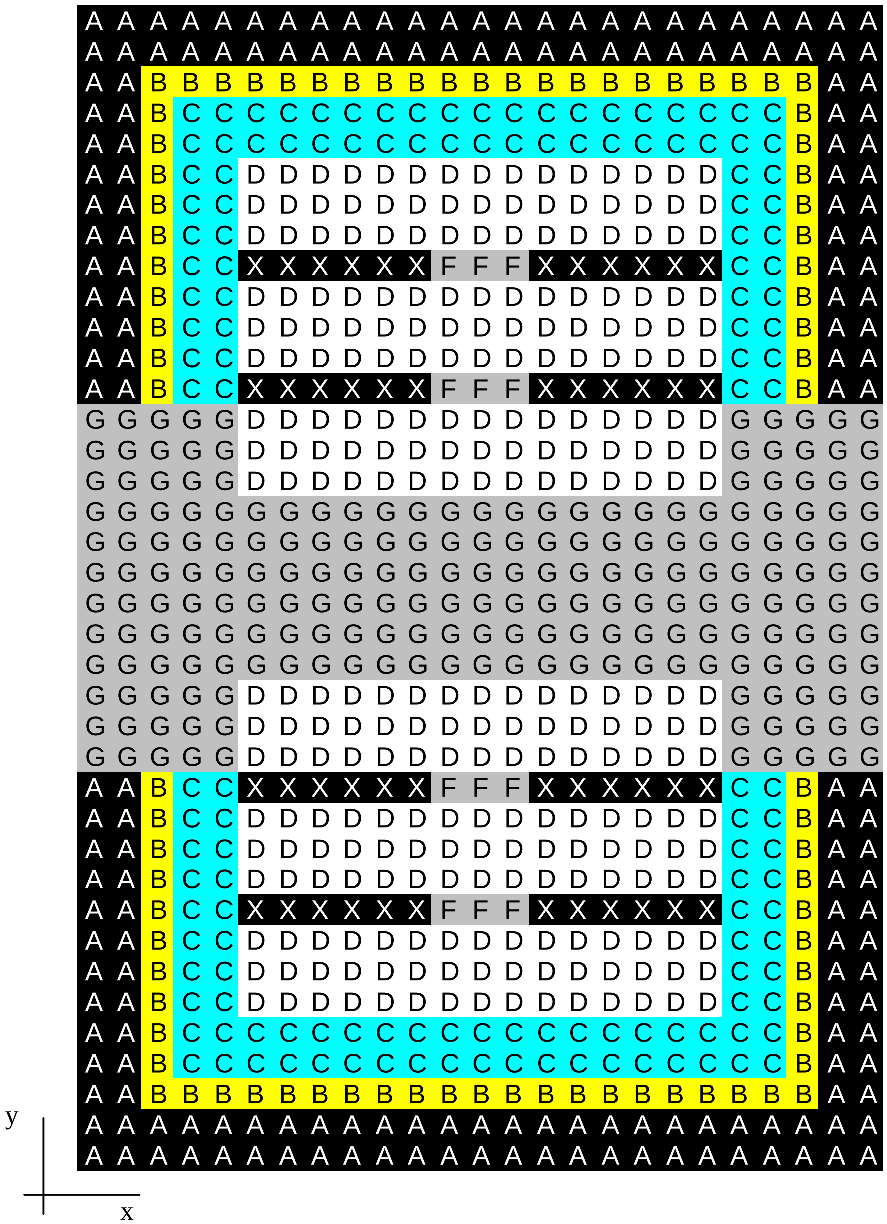

Figure 1: Core regions (A => reflector ; B => zircaloy; C => beryllium; D => fuel region; X => control material; F => aluminium guide; G => heavy water tank)

For purpose of transient analysis caused by the uncontrolled control bank withdrawal, we considered the disposition given by the figure 2, below.

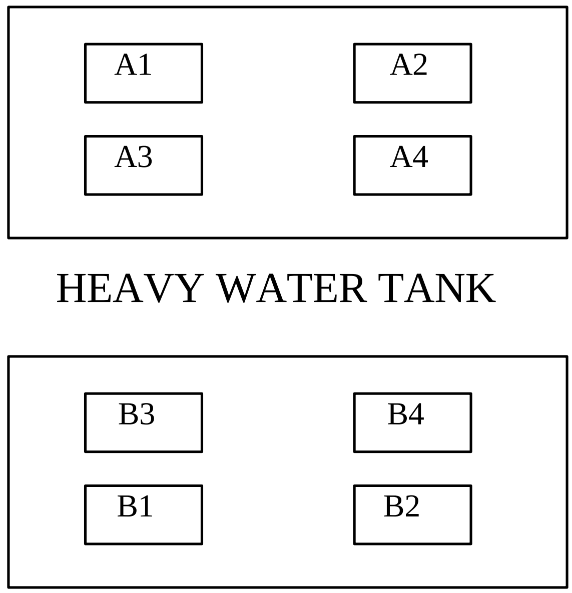

Figure 2: Control rods and their denominations.

With the denominations of the figure 2, the control rods form groups of control banks given by the table 2 following:

Table 2:

Control/Safety Banks	Control Rods
BC1 – Bank of Control 1	A1 + B1
BC2 – Bank of Control 2	A2 + B2
BC3 – Bank of Control 3	A3 + B3
BS – Bank of Safety	A4 + B4

During the driving of the control rods, the spaces left by them are filled out by water. With that, the following approach was used for calculation of the macroscopic constants:

$$CM = f.CM_{BA} + (1-f).CM_{H2O,}$$

where, in each energy group, one has:

CM = cross section, diffusion coefficient, etc.;
f = insertion fraction of control rods;
BA = index regarding to the absorber rod;
H2O = index regarding the water of completion of the space left by the absorber rod.

The macroscopic constants in the beginning of the transient, for the 4 energy groups, are generated by the HAMMER-TECHNION code /1,2/, considering the U_3Si_2 as the fuel material /3/. For the total fraction of delayed neutrons, the value 0.007 is used, relative to U235, whose constants of the groups of the precursors of delayed neutrons are shown in the table 3, following

Tabela 3: Delayed neutron constants.

i	β_i	λ_i (s^{-1})
1	2,66E-04	1,27E-02
2	1,49E-03	3,17E-02
3	1,32E-03	1,15E-01
4	2,85E-03	3,11E-01
5	8,97E-04	1,40E+00
6	1,82E-04	3,87E+00

<u>FEEDBACK EFFECTS</u>

During the transients, the capture macroscopic cross section in the fuel regions follows the resonance integral, for the g group, with dependence with the temperature T , in Kelvin, like this:

$$\Sigma_{cg}(T) = \Sigma_{cg}(300K)\left[1+\beta\left(\sqrt{T}-\sqrt{300K}\right)\right],$$

where,

$$\beta = c + d\left(\frac{S_F}{M_F}\right),$$

S_F = surface of the plate (cm^2),

M_F = fuel mass (g).

For the CRCN-R1 MultiPurpose Reactor, $S_F \,/\, M_F = 4.4$ cm^2/g . Using the values of Lamarsh/6 /, pg 459, for the dioxide of Uranium 238, one has: $c = 0.0061$ and $d = 0.0047$ .The scattering macroscopic cross section in the fuel region is in the way:

$$\Sigma_{sg \to g+1} = K \cdot \rho \ ,$$

where, K is an adjustment constant and ρ is the coolant density.

<u>TRANSIENT CHARACTERIZATIONS</u>

Three transients are analyzed in that work. One, simulating an accident in the startup of the reactor, other two simulating accidents with the reactor operating in power. The transients have begun with the reactor in the condition of critical state, and in that condition, the control rods are aligned in such a way that the fraction of insert of the banks of controls is given by $f = 0.31$, with the safety bank out of the core.

For lack of a more consistent methodology, applicable to research reactors, and besides, with the appreciable power production in CRCN-R1 MultiPurpose Reactor, we adopted the methodology of FSAR of Angra I /7 /. In that way, the parameters that limit the transients are: the trip points and the delays between the trip point and when the protectiom system iniciate the shutdown of the reactor, by the safety bank. In the table 4 it is exhibited the parameters that define the analyzed transients.

Table 4

Type	Setpoint	Withdrawal Speed (mm/s)	Delay (seg.)	$P/P_{NOMINAL}$
Fast Withdrawal of Control Bank in the Startup	35% $P_{NOMINAL}$	2.5	0.5	10^{-13}
Slow Withdrawal of Control Bank in the Power Operation	1^0C in the exit of the hottest channel	2.0	0.5	1.
Fast Withdrawal of Control Bank in the Power Operation	115% $P_{NOMINAL}$	2.5	0.5	1.

In a study of the fall times of the control rods of the Argonaut reactor, a reserach reactor of IEN (Nuclear Engineering Institute – CNEN/Brazil), one has found some values for the fall times that varied from 600 to 1800ms /8/. As the safety rods of the IEN reactor fall freely, outside of the tank of the core, it is of supposing that a rod falling in against coolant flow spends more time to be inserted, being consequently, submitted to a smaller acceleration than one gravity. Considering, then, the fall time of the safety rod in the calculations when admitting a value of 1000ms during the fall in the CRCN-R1 MultiPurpose Reactor shutdown, we have for an acceleration of fall of 140cm/s2, using the equation of the kinematics of the particle,

$$h_{BS} = \frac{1}{2} a_{BS} \cdot t_Q^2$$

where: h_{BS} is the safety rod length, 70cm; a_{BS} is the accelaration in the fall; t_Q is the fall time.

STARTUP ACCIDENT

The transient begins with the reactor operating in a power of 10^{-13} of the nominal when we simulated the accident removing the three control banks. In the instant in that the reactor power reaches the trip point of 35% of the nominal power, a sign is emitted to shut the reactor down, what it is made only by the fall of safety bank (BS). A time delay of 0.5 second is admitted between the sign of trip and the scram. The figure 3 shows the behaviors of the reactivity and the power fraction during 90s of transient. In the figure 4 is shown the temperatures in the center and in the surface of the plate with the largest power peak, the hottest channel.

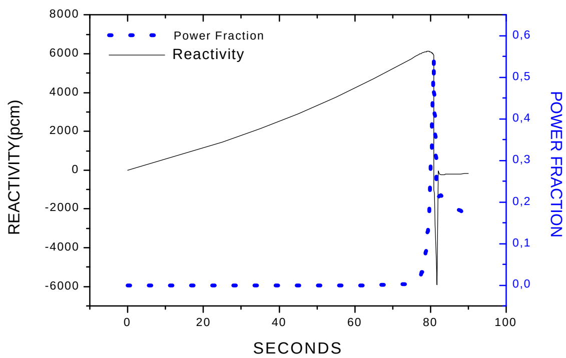

Figure 3: Reactivity and power fraction versus time.

It is noticed that the temperatures are below the limits of compromising of the physical integrity so much of the fuel, as of the cladding.

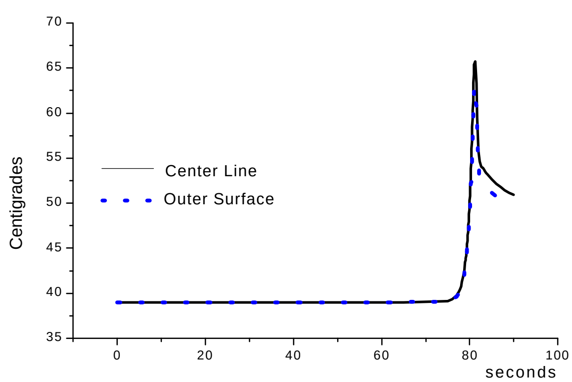

Figure 4: Temoeratures in the plate in te Starup Accident.

POWER OPERATION ACCIDENTS

The reactor is in a critical state, operating in the nominal power of 20 MW. Two transient types are calculated: one caused by a slow uncontrolled control rod withdrawal, and another representing a fast withdrawal. In both it is admitted a time delay of 0.5 second between the setpoint and the reactor scram.

a) SLOW UNCONTROLLED CONTROL ROD WITHDRAWAL

In that transient, the three control banks are retired at speed of 2.0 mm/s. The setpoint happens when the temperature of the coolant at the exit of the channel of the larger factor of power peak rises in 1^0C. Again, a sign is emitted to shut the reactor down, by the safety bank. The figure 5 shows the behaviors of the

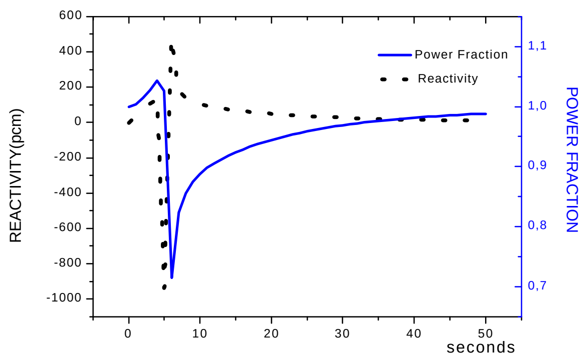

Figure 5: Reactivity and power fraction versus time.

reactivity and the reactor power fraction. They are also exhibited in the figure 6, the temperatures in the center line and at the outer surface of the plate of the channel. It is noticed that the temperatures, in that transient, are below the limits of compromising the physical integrity so much of the fuel, as of the cladding,

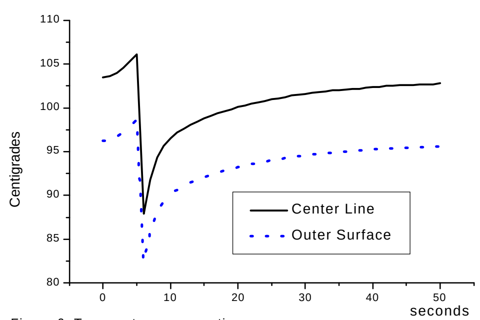

Figure 6: Temperatures versus time.

too. In the figure 7 is also shown the behavior of DNBR with the time.

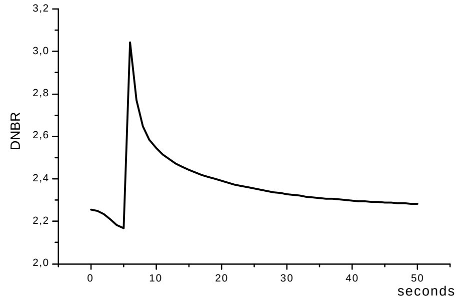

Figure 7: Estimate of DNBR versus time.

b) FAST UNCONTROLLED CONTROL ROD WITHDRAWAL

For simulating a fast control withdrawal, it is assumed a speed of 2.5 mm/s for the control bank BC3. The setpoint is due to 115% overpower. The reactor is also shutdown by the safety bank. The figure 8 show the reactor reactivity and the power fraction behaviors with the time. At the figure 9 is shown the

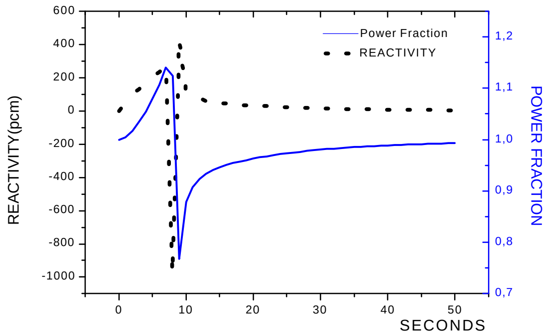

Figure 8: Reactivity and power versus time.

temperatures at the center and surface of the plate. However, in this simulation, the surface temperature exceeds the value of saturation temperature of the coolant, admitted to be at the atmospheric pressure. In that case the temperature limit would be of 100^{o}C. This result needs to be interpreted better, be for inadequate use of a feedback model in the calculation of the macroscopic nuclear constants, maybe by the nature of the overpower trip sign. Notice that in the slow transient, this situation is not verified, even with the simultaneous

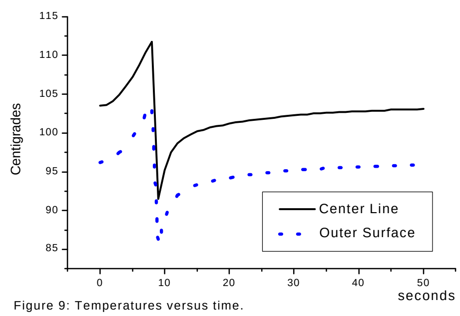

Figure 9: Temperatures versus time.

withdrawal of the 3 control banks. The figure 10 shows behavior of DNBR with the time, calculated by Bernath's correlation /9/.

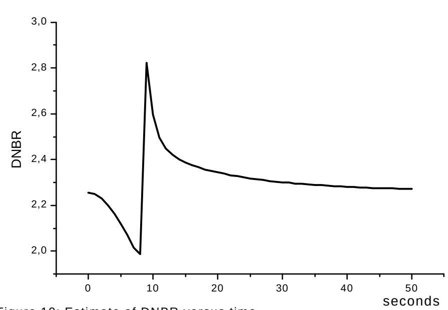

Figure 10: Estimate of DNBR versus time.

CONCLUSIONS

The reactor supports the simulate accidents well in startup and during a slow withdrawal with the reactor operating in the nominal power. It should be pointed out that the results were overestimated by the fact that it has admitted that the three control banks has been driven simultaneously in those transients. It is usual practice to allow only one bank to drive at a time.

The fast withdrawal of the control bank BC3 with the reactor operating in the nominal power showed that the thermal hydraulics limits should be reviewed. In that transient, in spite of a short time period, the temperature in the outer surface of a plate located in a position of larger power peak factor has surpassed the critical limit.

It was verified in the calculations the low effectiveness of safety bank. By the simulations they were verified its low efficiency to shutdown the reactor. In other words, it alone does not guarantee that the reactor remains subcritical indefinitely.

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