NEUTRONICS AND THERMALHYDRAULICS CHARACTERISTICS OF THE CANDU CORE FUELED WITH SLIGHTLY ENRICHED URANIUM 0.9% U_{235}

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Abstract

The interest concerning the slightly enriched uranium (SEU) fuel cycle is due to the possibility to adapt (to convert) the current reactor design using natural uranium fuel to this cycle. Preliminary evaluations based on discharged fuel burnup estimates versus enrichment and on Canadian experience in fuel irradiation suggest that for a 0.93% U-235 enrichment no design modifications are required, not even for the fuel bundle. The purpose of this paper is to resume the results of the studies carried on in order to clarify this problem. The calculation methodology used in reactor physics and thermal-hydraulics analyses that were performed adapted and developed the AECL suggested methodology. In order to prove the possibility to use the SEU 0.93% without any design modification, all the main elements from the CANDU Reactor Physics Design Manual were studied. Also, some thermal-hydraulics analyses were performed to ensure that the operating and safety parameters were respected. The estimations sustain the assumption that the current reactor and fuel bundle design is compatible to the using of the SEU 0.93% fuel.

1. Calculation methodology

The analysis methodology used in the studies regarding the neutron characterization of the SEU 0.9 % fueled core represents an adaptation and development of similar methods used by AECL for this kind of analyses. It can be considered that the principal aspects characterizing this type of reactor which influence the choice of calculation methods are:

- utilization of heavy water as moderator;
- separation of moderator from coolant;
- horizontal fuel channels and perpendicular reactivity devices;
- continuous refueling.

By the reduced moderator absorption, the PHW lattices are well moderated. This makes possible that the neutron balance be well approximated in the four factors formula, and the neutron flux be well approximated in the Westcott formalism.

Subsequently, the neutron parameters are built most often in the age-diffusion approximation. For natural Uranium, one uses the POWDERPUFS (PPV) computer code (ref. /1/). For detailed and more rigorous information, one resorts to the LATREP code (ref. /2/), which deals with the multigroup flux distribution by collision probabilities. Though these two codes are the most used programs for the CANDU reactor design analysis, one uses the WIMS code too for the multigroup transport calculation in some situations (ref. /3/).

For higher values of U_{235} content in the fuel some questions rise for the PPV results just because these results are based on the experimental data obtained for natural Uranium. That is why one uses especially the LATREP and WIMS computer codes for determining the cell parameters in the case of SEU fuel cycle. The flux distribution determination is done by two group, 3D diffusion calculations. The tridimensional model is required by the geometrical configuration with horizontal channels and perpendicular reactivity devices. An important feature is the existence of a neutron properties distribution induced by the presence of a different burnup fuel distribution as a result of a continuous bidirectional refueling. These aspects are taken into account by the way of building the macroscopic constants used in the diffusion calculations, i.e.:

- 'homogeneous calculations', which treat both global and local neutron balance, considering a uniform distributed fuel with averaged properties, up to the discharge burnup;

- 'time average calculations', which corespond to a similar method distinguished by the fact that one considers average compositions that differ along the channel, the time average being done up to irradiations depending on channel power and residence time, in compliance with the refueling scheme;

- 'instantaneous calculations', which treat the problem by appointing each bundle properties calculated at the burnup that results from the history of irradiation.

The 'instantaneous' constants approximation is used especially for the refueling simulation or for studies regarding the refueling strategies.

For advanced fuel cycles, the SERA program was developed, a program similar to FMDP, ref. /4/. The SERA program, ref. /5/, has a modular structure in which the macroscopic constants can be automatically generated with the LATREP code or can be taken from a WIMS built file. It can be used in all the three aproximations mentioned above. The code provides the possibility of building the constants in function of the real irradiation history for each bundle. In the multigroup diffusion approximation the reactivity devices are represented by specific homogenized cross sections, based on proper local calculations.

The PIJXYZ code was used. This integral transport program solves the multigroup equations in the whole supercell volume (ref. /6/). The needed material properties, of the fuel, pressure tube, reactivity device are determined by the WIMS program.

2. Cell parameters and reactivity effects

The neutron balance assessment at the cell level enables us to evaluate the reactivity effects produced by the variations of different parameters. These reactivity effects underlie the requirements for the different reactivity devices and, by comparison with similar natural Uranium effects, one can know if it is necessary to change them.

The reactivity effects for SEU 0.9 % were analyzed by E. Nichita, for thermalhydraulics parameters variations in ref. /7/ and by P. Laslau for the Xenon concentration variation in ref. /8/.

The multiplication factor evolution with burnup for SEU 0.9 % is shown in figure no.1. The comparison with natural Uranium shows that, obviously, it starts from a higher initial k value and the Plutonium peak does not appear. This can be explained by the higher fissile concentration that makes the variation in the multiplication factor due to U_{235} exhaustion exceed the one due to the Pu build-up.

In figure no. 2 the integrated multiplication factor evolution is shown, a parameter that can be linked with the global balance in the hypothesis that the bundles in the core are characterized by a uniform irradiation distribution between zero and the discharge irradiation. This interpretation permits the discharge burnup assessment, considering a reactivity need for the neutron leak and reactivity devices and incidental absorption compensation. For example, in figure no. 2, one can estimate a discharge burnup of 14983 Mwh/t, resulting from the intersection between the multiplication factor evolution curve and the 47 mk line that corresponds to the assessed reactivity need.

In figure no. 3 the radial power distribution evolution for average bundle is shown; this evolution is the result of superimposing the effects due to changes in fissile isotopes concentrations and in flux microdistribution.

The moderator temperature reactivity effect is shown in figure no. 4, in which the multiplication coefficient variation with the moderator temperature for SEU 0.9 % is presented.

The fuel temperature reactivity effect is shown in figure no. 5, in which the multiplication coefficient variation with the fuel temperature for SEU 0.9 % is presented.

The void reactivity effect is shown in figure no. 6, in which the multiplication coefficient variation with the coolant void for SEU 0.9 % is presented.

The SEU 0.9 % and natural Uranium reactivity effects comparison shows the multiplication factor variations have the same behavior regarding each perturbation parameter and the same evolution trend with irradiation even if, for fuel at equilibrium, there are some small differences.

The reactivity effects due to Xenon have a special importance both in the steady state, by its large reactivity, and while operating, by the important neutron balance variations, that are induced at passing from one state to another and by the associated transients.

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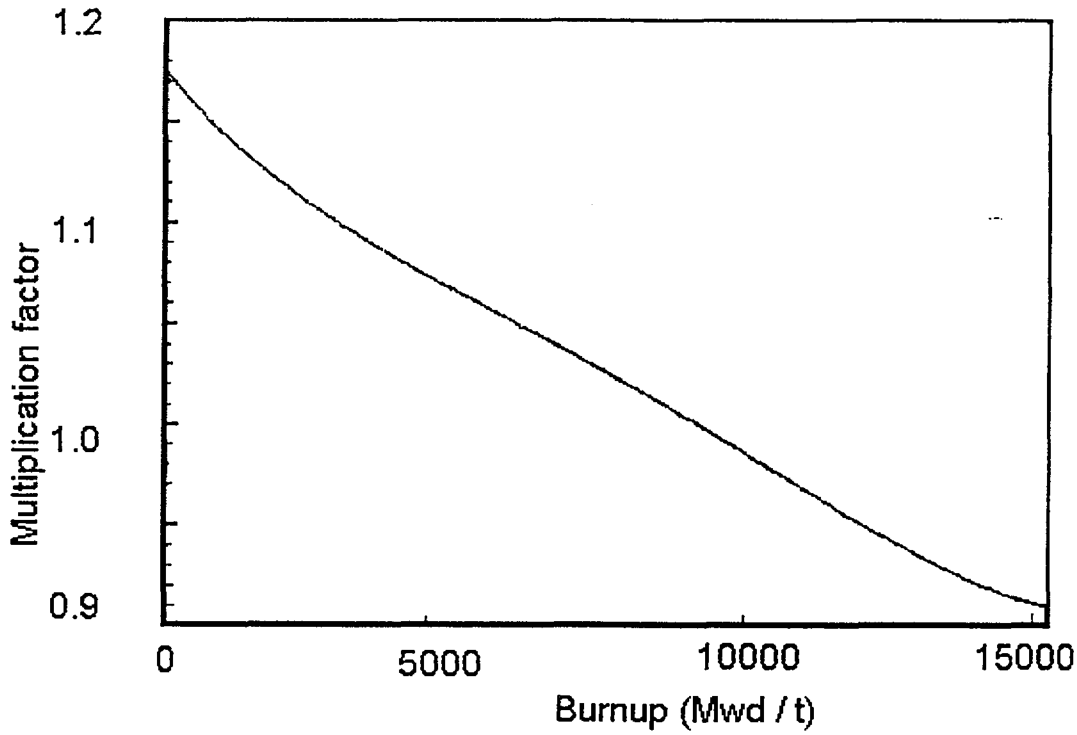

Figure 1. Multiplication factor versus burnup

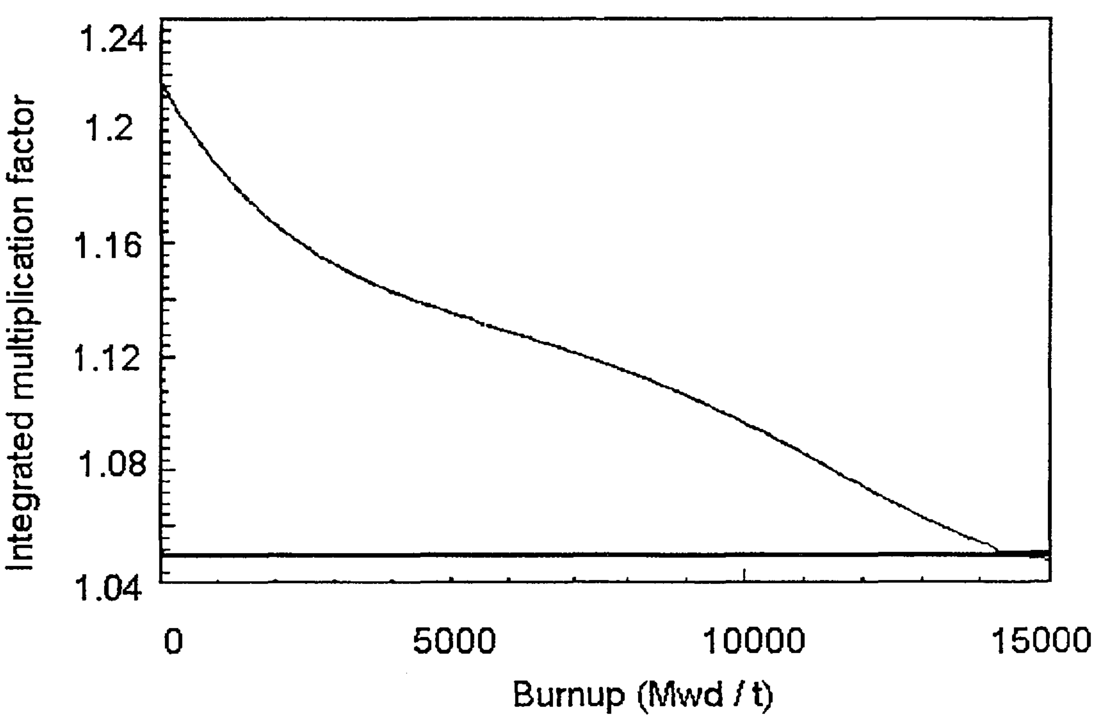

Figure 2. Integrated multiplication factor versus burnup

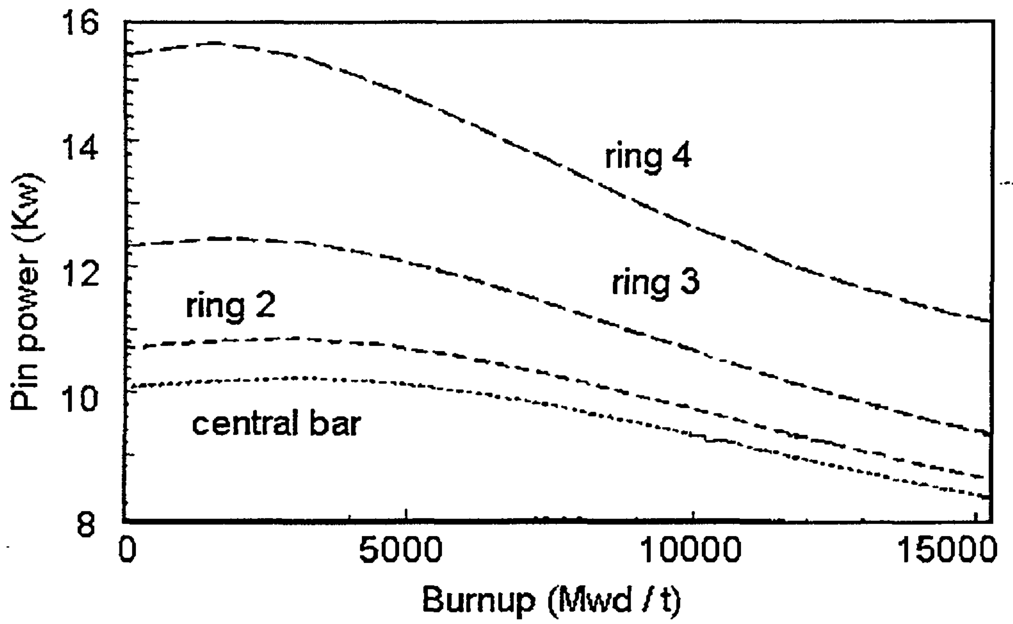

Figure 3. Pin power versus burnup

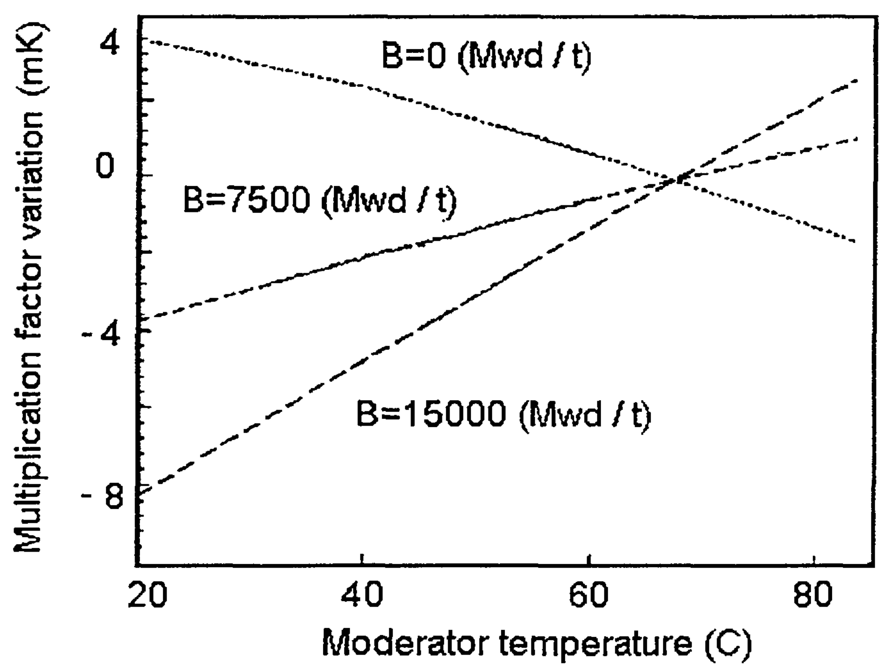

Figure 4. Multiplication factor variation versus moderator temperature

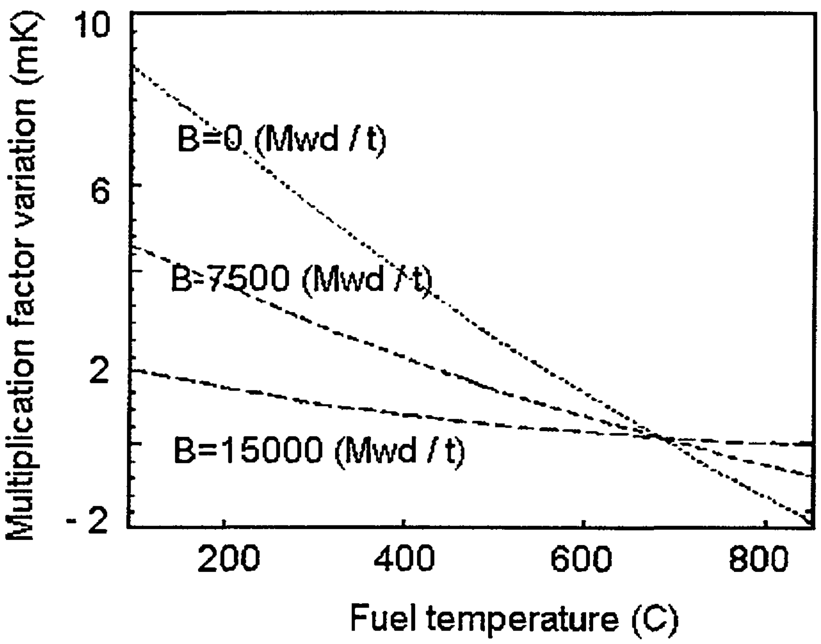

Figure 5. Multiplication factor variation versus fuel temperature

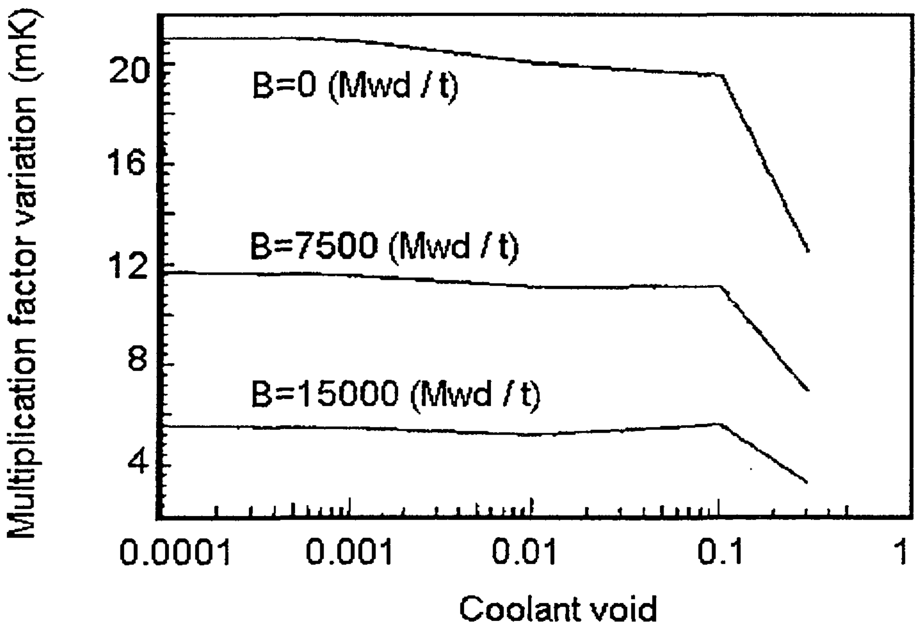

Figure 6. Loss of coolant reactivity effect

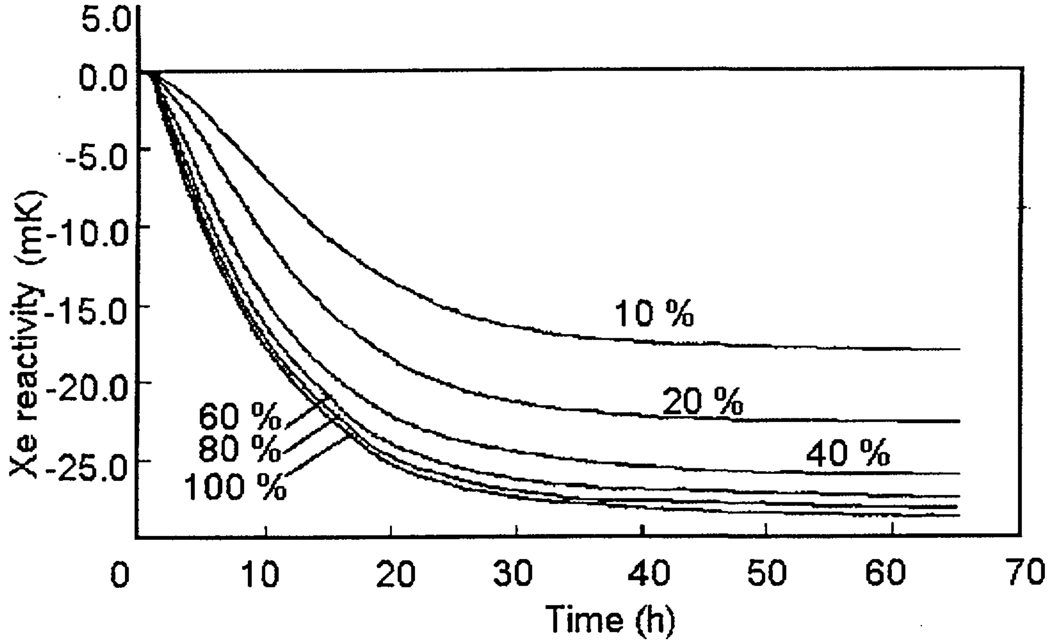

Figure 7. Xe reactivity transients after startup

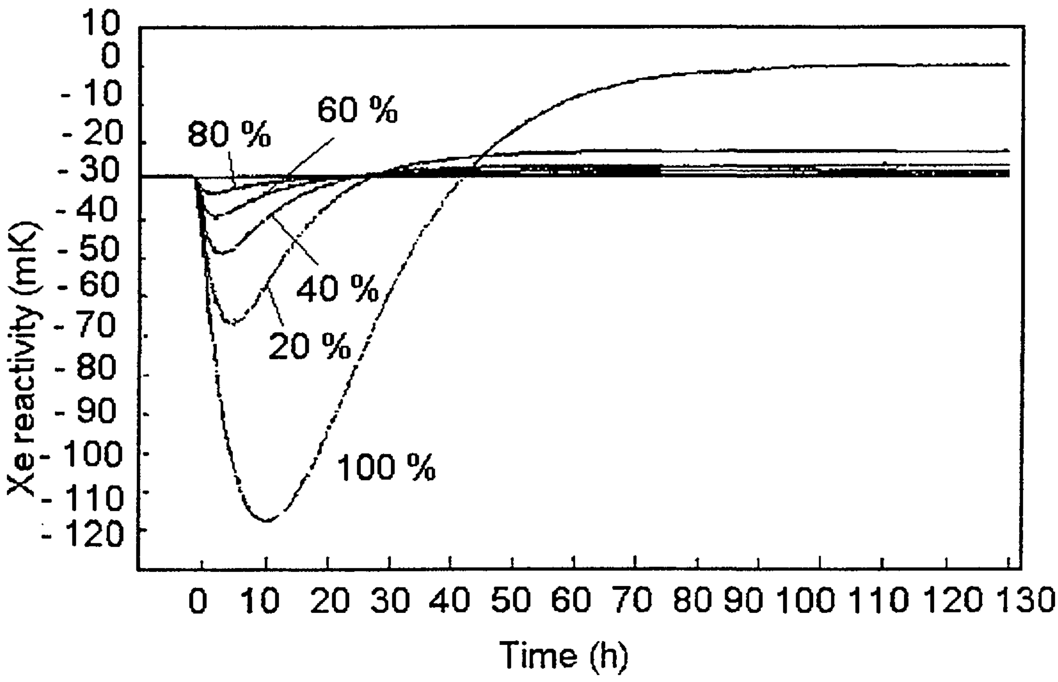

Figure 8. Xe reactivity transients after reduction of power from 100 %

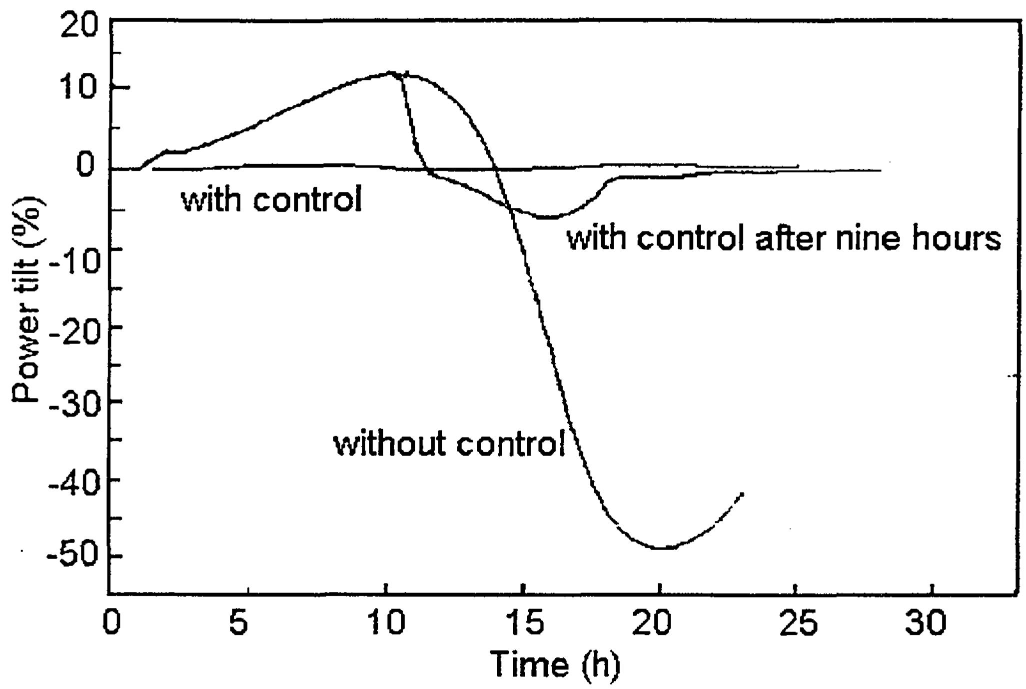

Figure 9. First azimuthal mode evolution

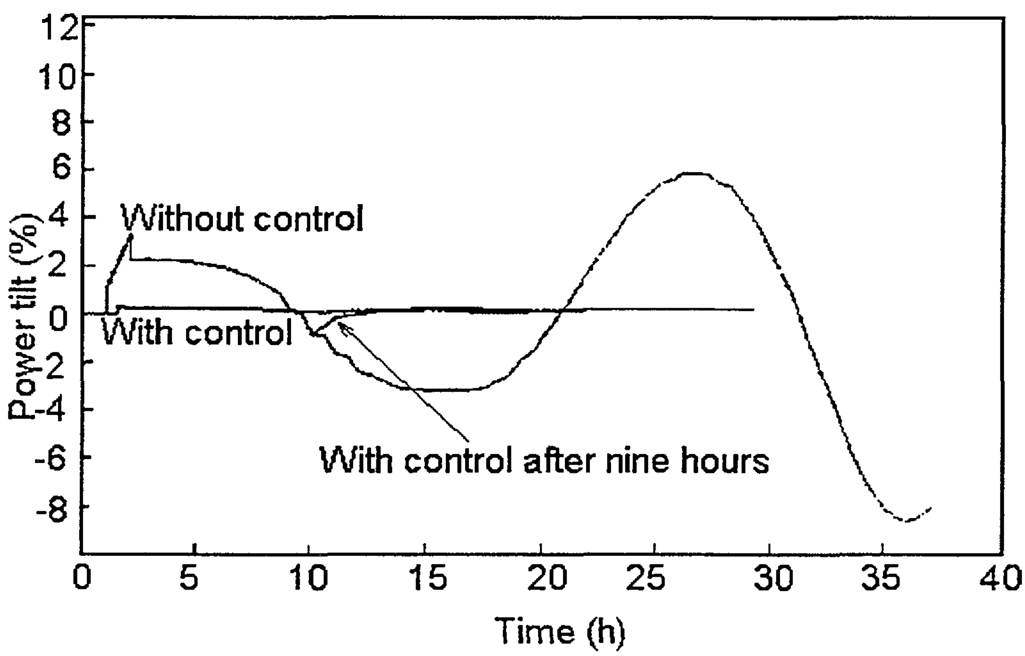

Figure 10. First axial mode evolution

The cumulated yield is over 7 %, for SEU fuel, from which over 10 % is prompt component. By production and loss of Xenon, by absorption and desintegration, a equilibrium state is reached, that, for SEU fuel, coresponds to a reactivity of 28-29 mk at nominal power. This can be identified in figure no. 7, where the Xe transients at startup for different power levels are shown.

In figure no. 8, the Xe reactivity transients resulted by the reduction of power up to different levels are shown, starting from nominal value. These transients are generated by the mismatch between reduction and production, whose decrease is delayed by Iodine desintegration. For that reason, in the beginning, a Xe concentration increase is taking place, and consequently, its negative reactivity increases, followed by a decrease to the equilibrium value given by the power value on each level.

Another important aspect that needs attention is the stability of the core with respect to the local Xe transients. In the PHW reactors, usually very large, with a strong flux flattening and relatively high power densities, Xe density spatial distribution oscillations can be observed. For setting the conditions in which these oscillations could appear and if they would naturally diminish or increase, special analyses are required to link the different Xe distribution spatial forms and the transient mode corresponding to them.

In this case, one was able to avoid the mentioned analyses, having a set of conclusions of such analyses for the natural Uranium core. In fact, one analyzed the transients of the first two modes, in the instability order, identified for natural Uranium. This procedure is justified by the fact that the Xe effects look alike for the two fuel types.

The results obtained for SEU fuel are gathered in figures 9 and 10, where the relative deformations of the power distribution characterizing these two modes, respectively left-right and front-back, are shown. As a general observation, one can state that the SEU fuel core is more unstable at the Xe oscilations, especially for the axial modes due to the additional axial flattening.

3. Equilibrium power distribution and refueling strategies

For SEU fueled core, due to the sharper channel neutron properties variations, the time average approximation calculations become the base of simulation and not only an improvement of the results obtained by homogeneous calculations. Because the average time calculations imply handling scheme elements, the equilibrium power distribution and refueling strategy are settled at the same time. The SEU utilization implies new refueling schemes, especially due to the sharper decrease with burnup and to larger power deformations after refueling.

There are some options which were identified and presented in scientific literature, for example ref. /9/ and /10/.

For choosing a convenient scheme, a large option range was analyzed. These analyses were conducted on a 7x7 channels lattice, watching the residence time and

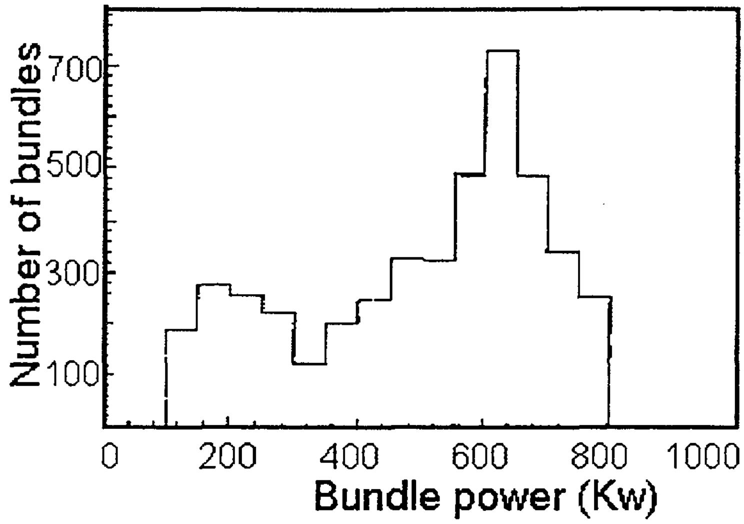

Figure 11. Number of bundles for different power intervals

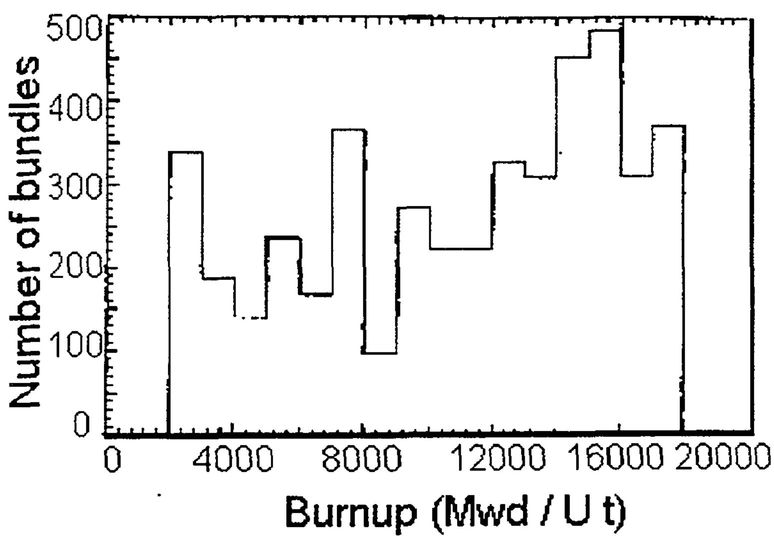

Figure 12. Number of bundles for different burnup intervals

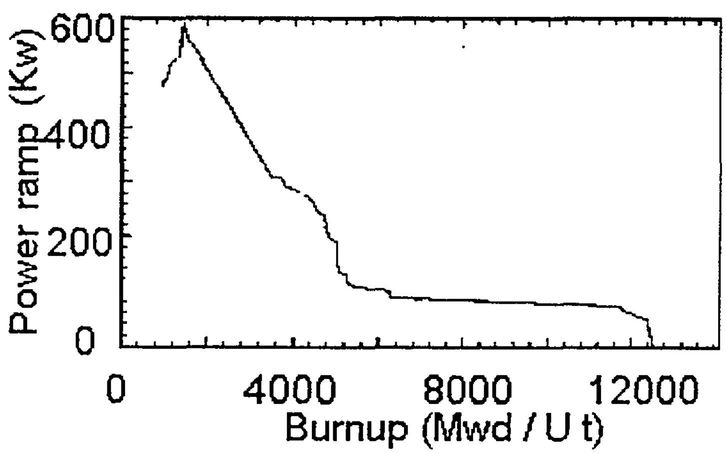

Figure 13. Power increase envelope during refueling power ramps
(time average calculations)

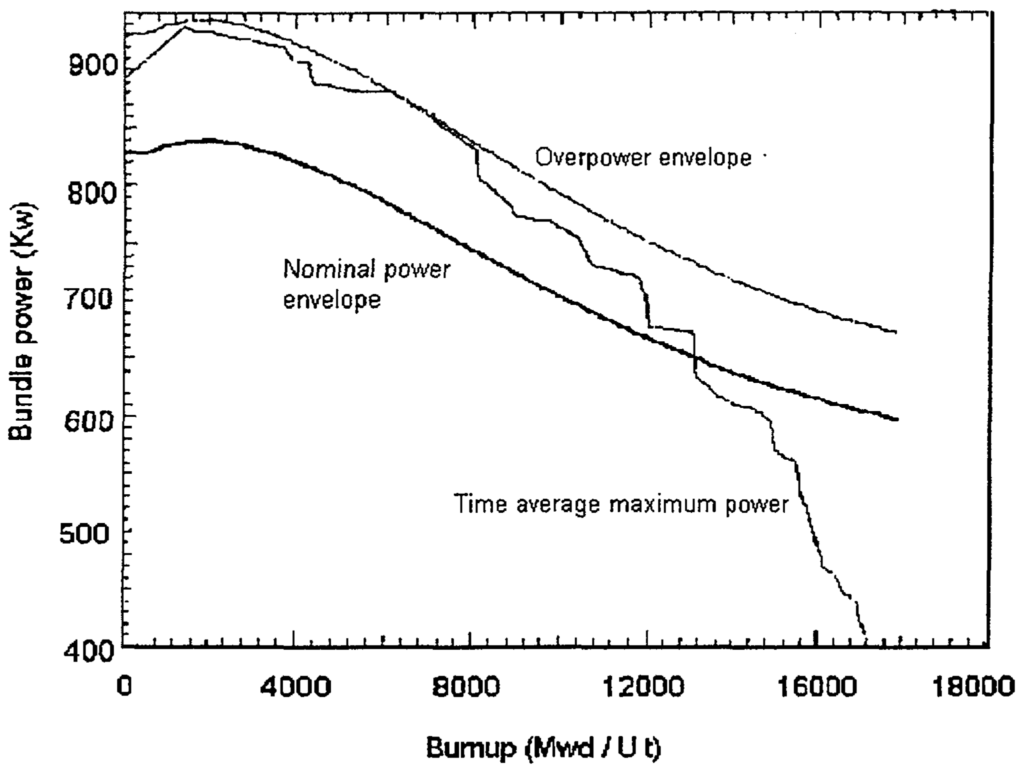

Figure 14. Power envelopes

the requirement conformation for different axial handling schemes. Among these schemes, to be analyzed at the core level, the following axial handling schemes were selected:

- 4 bundles;
- 4+2 bundles;
- 2 bundles.

These were analyzed with regard to a parameter series which define the core configuration, for example:

- burnup zones number;
- burnup zones dimensions;
- axial shift scheme

and the restrictions imposed by the maximum channel and bundle powers.

The tridimensional diffusion calculations were coupled with thermalhydraulic analyses for selecting an adequate configuration set. These have the role of providing critical channel power estimations for the chosen axial power distribution. For that, one can use as a refueling scheme selection criterion a maximum ratio between the critical and nominal channel powers.

The refueling scheme which proves convenient consists in feeding the central zone with 2 bundles and the external zone with 4 bundles. For this scheme, one have 3 burnup concentric zones on which one imposes different discharge burnup degrees.

TABLE 1. BURNUP ZONES MAP

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									1	1	1	1	1	1								
B						1	1	1	1	1	1	1	1	1	1	1	1					
C					1	1	1	1	1	3	3	3	3	1	1	1	1	1				
D				1	1	1	3	3	3	3	3	3	3	3	3	3	1	1	1			
E			1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1		
F			1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1		
G		1	1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1	1	
H		1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1	
J	1	1	1	3	3	3	3	3	2	2	2	2	2	2	3	3	3	3	3	1	1	1
K	1	1	1	3	3	3	3	3	2	2	2	2	2	2	3	3	3	3	3	1	1	1
L	1	1	1	3	3	3	3	3	2	2	2	2	2	2	3	3	3	3	3	1	1	1
M	1	1	1	3	3	3	3	3	2	2	2	2	2	2	3	3	3	3	3	1	1	1
N	1	1	1	3	3	3	3	3	2	2	2	2	2	2	3	3	3	3	3	1	1	1
O	1	1	1	3	3	3	3	3	2	2	2	2	2	2	3	3	3	3	3	1	1	1
P		1	1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1	1	
Q		1	1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1	1	
R			1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1		
S			1	1	1	3	3	3	3	3	3	3	3	3	3	3	3	1	1	1		
T				1	1	1	3	3	3	3	3	3	3	3	3	3	1	1	1			
U					1	1	1	1	1	3	3	3	3	1	1	1	1	1				
V						1	1	1	1	1	1	1	1	1	1	1	1					
W									1	1	1	1	1	1								

TABLE 2. REFUELING SCHEME

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									4	-4	4	-4	4	-4								
B						4	-4	4	-4	4	-4	4	-4	4	-4	4	-4					
C					4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4				
D				4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4			
E			4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4		
F			-4	4	-4	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	4	-4	4		
G		-4	4	-4	4	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-4	4	-4	4	
H		4	-4	4	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-4	4	-4	
J	4	-4	4	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-4	4	-4
K	-4	4	-4	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	4	-4	4
L	4	-4	4	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-4	4	-4
M	-4	4	-4	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	4	-4	4
N	4	-4	4	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-4	4	-4
O	-4	4	-4	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	4	-4	4
P		-4	4	-4	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	4	-4	4	
Q		4	-4	4	-4	2	-2	2	-2	2	-2	2	-2	2	-2	2	-2	4	-4	4	-4	
R			4	-4	4	-2	2	-2	2	-2	2	-2	2	-2	2	-2	2	-4	4	-4		
S			-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4		
T				-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4			
U					-4	4	-4	4	-4	4	-4	4	-4	4	-4	4	-4	4				
V						-4	4	-4	4	-4	4	-4	4	-4	4	-4	4					
W									-4	4	-4	4	-4	4								

TABLE 3. REFUELING SCHEME AND BURNUP DISTRIBUTION

BURNUP ZONE	1	2	2	3
AXIAL REFUELING SCHEME	2	2	4	4
BURNUP(MWD/T)	15410	16050	16050	13530
AVERAGE BURNUP(MWD/T)	15050			

TABLE 4. CHANEL POWER DISTRIBUTION (KW)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									3408	3494	3568	3568	3494	3408								
B						3137	3719	4211	4475	4626	4669	4669	4626	4475	4212	3720	3138					
C					3453	4125	4718	5131	5362	5274	5248	5249	5275	5363	5132	4719	4127	3454				
D				3602	4357	4996	5299	5585	5734	5751	5647	5647	5751	5735	5586	5300	4998	4359	3603			
E			3529	4453	5140	5455	5772	5959	6025	6001	5877	5878	6002	6026	5960	5774	5457	5142	4454	3531		
F			4356	5193	5698	5836	6009	6099	6050	6009	5948	5948	6010	6051	6101	6011	5838	5700	5195	4358		
G		3971	4959	5711	5977	5988	6132	6186	6119	6098	6117	6117	6099	6120	6188	6133	5990	5980	5713	4961	3973	
H		4501	5455	6043	5961	6066	6157	6203	6179	6182	6220	6221	6183	6180	6204	6159	6068	5873	6046	5457	4503	
J	3651	4855	5794	6053	6065	6124	6179	6224	6228	6295	6311	6312	6296	6290	6225	6181	6127	6067	6055	5796	4857	3652
K	3859	5130	6029	6204	6137	6163	6224	6257	6316	6291	6217	6218	6291	6317	6259	6226	6165	6140	6207	6032	5132	3861
L	3991	5273	6164	6328	6306	6310	6293	6291	6330	6282	6176	6177	6283	6332	6293	6296	6313	6308	6331	6167	5276	3993
M	3993	5279	6181	6364	6369	6373	6329	6307	6336	6283	6176	6176	6284	6337	6308	6331	6375	6371	6367	6183	5282	3994
N	3861	5141	6067	6296	6342	6364	6317	6292	6326	6291	6216	6216	6292	6327	6294	6319	6367	6345	6299	6069	5143	3862
O	3644	4854	5815	6127	6254	5310	6254	6243	6286	6289	6306	6306	6290	6288	6245	6256	6312	6257	6129	5817	4856	3645
P		4470	5422	6024	5971	6076	6139	6169	6147	6162	6212	6212	6162	6149	6171	6140	6078	5973	6026	5424	4472	
Q		3904	4862	5577	5812	5829	6000	6083	6045	6061	6114	6114	6061	6046	6084	6002	5831	5814	5579	5864	3905	
R			4205	4960	5347	5502	5772	5927	5928	5945	5974	5974	5946	5929	5928	5774	5503	5349	4962	4207		
S			3365	4193	4752	5081	5499	5737	5847	5863	5777	5777	5863	5848	5738	5490	5082	4753	4195	3366		
T				3385	4083	4708	5032	5346	5516	5542	5437	5437	5543	5517	5347	5033	4709	4084	3386			
U					3256	3909	4481	4894	5122	5023	4944	4944	5024	5123	4895	4482	3910	3256				
V						2974	3532	4008	4260	4393	4426	4426	4393	4260	4009	3533	2975					
W									3239	3316	3389	3389	3316	3239								

The burnup zone distribution is shown in table no. 1 and the distribution of the regions in which the shift is done with 2, respectively with 4 bundles is shown in the table no. 2. One can see the burnup zones limits do not correspond to those of the different handling zones limits, in the second zone being both 2 bundles fed channels and 4 bundle fed channels. This aspect comes even more clearly from table no. 3, in which the discharge burnup on the 3 regions are shown.

The principal elements that characterize the equilibrium core, considering this refueling scheme, are presented in table no. 4, which contains the channel power distribution, and in figures from 11 to 14, as follows:

- the bundles number distribution versus power intervals (figure no. 11);
- the bundles number distribution versus burnup (figure no. 12);
- power increase envelope during refueling power ramps (figure no. 13);
- power envelope (figure no. 14).

For setting the fuel bundle operating requirements the instantaneous power distribution is essential. To obtain the data required for fuel behavior analysis the core operation and refueling were simulated. These simulations results are reported by I. Patrulescu in ref. /11/. The calculations started from an instantaneous situation generated by the random age method, simulating the refueling on a 200 days interval

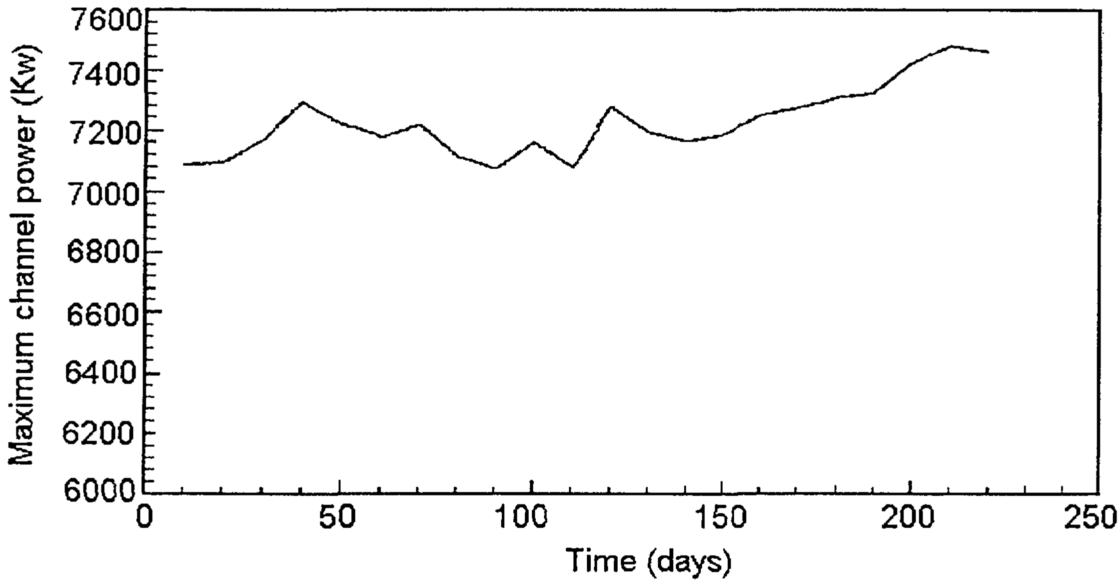

Figure 15. Maximum channel power refueling transient

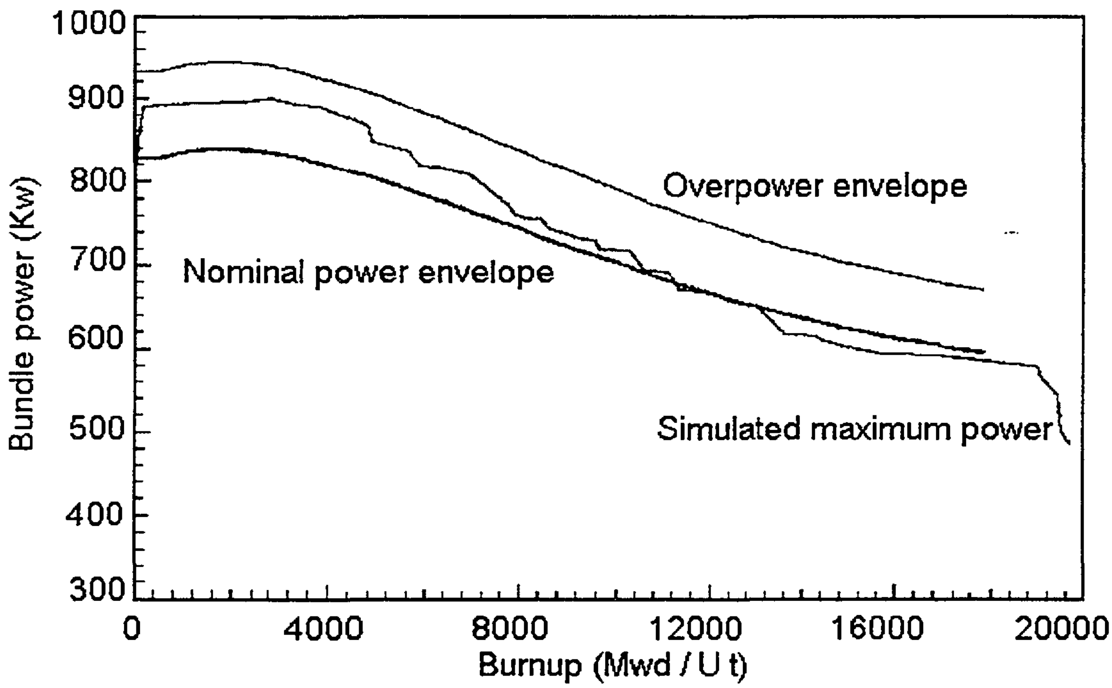

Figure 16. Bundle power envelopes (refueling simulation)

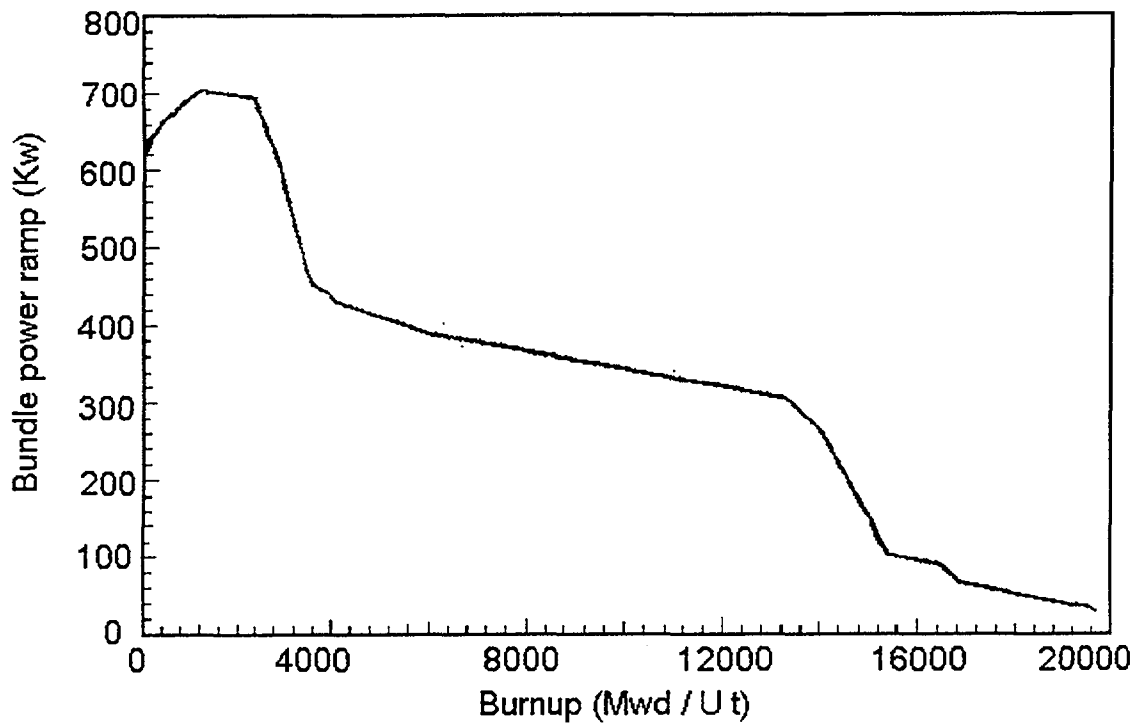

Figure 17. Power increase envelope during refueling power ramps

TABLE 5. DRY OUT CRITICAL CHANEL POWER DISTRIBUTION (KW).

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									5825	6112	5845	5845	5285									
B						5697	5992	7043	7311	7347	7669	7669	7346	7311	7043	5992	5697					
C					5993	6471	7347	7757	8049	8414	8498	8498	8414	8049	7757	5347	6471	5993				
D				6046	6815	7593	7945	7644	8853	8977	9041	9041	8976	8853	7644	7946	7593	6815	6046			
E			6332	6773	7830	8319	8678	8901	9103	9189	9454	9454	9189	9103	8901	8678	8319	7830	6773	6332		
F			6913	7739	8457	8823	9000	9183	9148	9235	9194	9194	9235	9148	9183	9000	8823	8457	7739	6913		
G		6509	7522	8176	8790	9195	9112	9209	9163	9157	9221	9221	9157	9163	9209	9112	9125	8790	8176	7522	6509	
H		7173	8022	8761	9086	9173	9130	9208	9142	9089	9096	9096	9089	9139	9201	9120	9173	9086	8772	8021	7173	
J	5979	7387	8370	8984	9380	9313	9313	9321	9170	9199	9098	9098	9199	9170	9321	9315	9312	9380	8984	8370	7387	5979
K	6176	7581	8488	9180	9437	9248	9282	9289	9201	9145	9043	9042	9145	9201	9289	9282	9283	9437	9180	8488	7581	6176
L	6535	7565	8636	9169	9559	9077	9400	9087	9240	8799	8880	8581	9110	8928	9395	9093	9371	9288	9323	8458	7754	6289
M	6285	7715	8442	9209	9241	9482	9064	9352	9086	9115	8692	9008	8802	9376	9036	9379	9176	9481	9050	8622	7515	6531
N	6438	7659	8496	9126	9449	9378	9274	9402	9216	9136	8970	8970	9136	9216	9402	9274	9378	9449	9126	8496	7659	6438
O	6321	7384	8396	8948	9295	9269	9313	9324	9221	9112	8983	8991	9112	9221	9324	9313	9269	9295	8948	8397	7384	6321
P		7284	8307	8749	9188	9307	9265	9340	9245	9188	9083	9083	9188	9246	9340	9265	9307	9188	8749	8307	7284	
Q		6418	7419	8527	9151	9204	9165	9209	9254	9286	9199	9199	9286	9254	9290	9165	9204	9151	8527	7419	6418	
R			6970	7916	8702	9334	9059	9258	9257	9266	9269	9269	9266	9257	9258	9059	9334	8702	7916	6970		
S			5891	7107	7915	8567	9062	8996	9239	9484	9530	9530	9484	9239	8996	9062	8567	7915	7107	5982		
T				6118	6813	7717	8298	8935	8905	9032	9079	9079	9033	8905	8935	8299	7717	6813	6118			
U					5737	6319	7302	8362	8405	8805	8820	8820	8805	8406	8362	7302	6319	5737				
V						5356	6013	6835	7011	7259	7280	7280	7259	7011	6835	6013	5356					
W									5862	5982	6226	6227	5981	5862								

with a 10 days step. The refueled channel determination was done off-line, following the procedure and the refueling rules for natural Uranium. A 4 % deviation from the adequate value was admitted for the maximum channel power, based on the previous NUCCP thermalhydraulics analyses of relative differences of critical power (dry out). In figures no. 15 to 17 the results of these simulations are given, respectively:

- maximum channel power refueling transient (figure no. 15);
- bundle power envelope (figure no. 16);
- power increase envelope during refueling power ramps (figure no. 17);

TABLE 6. FUEL MELTING CRITICAL CHANEL POWER DISTRIBUTION (KW).

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22
A									7121	7449	7298	8299	7449	7121								
B						6822	7445	7747	8163	8080	8274	8274	8081	8163	7747	7445	6822					
C					6940	7692	8069	8486	8593	8795	8741	8741	8795	8593	8846	8069	7691	6941				
D				6990	7829	8218	8653	8304	9199	9193	9173	9173	9193	9199	8304	8653	8218	7829	6990			
E			7153	7781	8291	8767	9112	9425	9582	9635	9565	9565	9635	9582	9425	9112	8767	8291	7780	7154		
F			7711	8249	8704	8986	9504	9861	10020	10111	9952	9952	10111	10020	9861	9503	8986	8703	8249	7711		
G		7444	8167	8550	8910	9160	9730	10009	10397	10523	10373	10373	10523	10397	10009	9730	9160	8910	8550	8167	7444	
H		7888	8467	8931	9086	9341	9884	10263	10475	10422	10474	10474	10422	10475	10263	9884	9341	9086	8931	8466	7888	
J	7211	8155	8686	9054	9380	9615	10305	10702	10562	10596	10518	10518	10596	10562	10702	10305	9615	9380	9054	8686	8154	7212
K	7427	8292	8823	9298	9570	9946	10503	10672	10577	10528	10479	10479	10528	10578	10672	10503	9946	9570	9298	8823	8292	7426
L	7568	8852	8933	9538	9871	10785	10675	10476	10656	10289	10386	10329	10488	10348	10803	10741	10164	9986	9473	9074	8399	8266
M	8273	8407	9086	9403	10045	10300	10639	10759	10321	10497	10305	10450	10281	10828	10464	10677	10921	9910	9544	8944	8916	7570
N	7547	8348	8869	9323	9815	10181	10532	10828	10618	10527	10378	10377	10527	10618	10828	10532	10181	9815	9323	8869	8348	7547
O	7359	8177	8738	9104	9482	9831	10338	10719	10653	10489	10390	10390	10489	10553	10718	10338	9831	9482	9104	8738	8177	7359
P		7983	8575	8978	9188	9475	10025	10348	10620	10581	10469	10468	10582	10620	10348	10025	9475	9188	8978	8575	7983	
Q		7417	8102	8658	9151	9204	9747	10088	10465	10615	10400	10400	10615	10465	10088	9747	9204	9151	8658	8102	7417	
R			7717	8242	8702	9334	9449	9884	10118	10160	10049	10049	10160	10118	9884	9449	9334	8702	8242	7717		
S			7018	7827	8133	8715	9250	9453	9680	9818	9693	9693	9818	9680	9453	9250	8715	8133	7827	7018		
T				6996	7786	8186	8713	9135	9200	9229	9235	9235	9229	9200	9135	8713	8186	7786	6996			
U					6843	7646	8033	8692	8703	8977	8853	8853	8977	8703	8692	8033	7646	6843				
V						6709	7460	7647	7970	7995	8039	8040	7995	7970	7647	7460	6708					
W									7149	7403	7419	7419	7404	7149								

4. SEU fuel thermalhydraulics conditions evaluations

These evaluations were conducted only for defining the requirements imposed by the thermalhydraulics conditions and estimating the compliance with these conditions. The primary circuit characteristics are considered unchanged with respect to those for natural Uranium, the principal change reffering to both the radial and especially axial core power distributions. To perform these analyses the NUCCP code was used, a code used by AECL for designing the primary circuit and setting the operation conditions (ref. /12/). In the first phase, the global circuit characterization calculations were done. For this geometry one determines the thermalhydraulic parameters distribution in the entire circuit, especially the pressure drops and flows. The results were very close to those obtained for natural Uranium, the differences being not essential.

In the second phase, analyses were conducted channel by channel, considering the circuit part between the two headers in detail and imposing boundary conditions, respectively maintaining unchanged the pressure drop between the two headers. For each channel, the detailed thermalhydraulic parameters distribution was determined, for example: temperatures, pressure drops, flows, title and critical power (dry out and melting).

In ref. /13/, L. Bratu estimated, by this method, the thermalhydraulic parameters for SEU 0.9 % fueled core. These critical powers distributions are shown in the table no. 5 (for dry out) and table no. 6 (melting). The importance of these assessments consists in the fact that they permit a global and rapid evaluation for the safety requirements compliance. One can see the critical ratios (the ratio between the critical and the nominal channel powers) are, generally, higher for SEU than for natural Uranium. This is due to the axial channel distribution asymetry, with the maximum shifted to to the channel inlet and having lesser values at the channel outlet, where the conditions of dry out are more rapidly created. This constitutes an important advantage, allowing an increased flexibility in reactor operation by larger margins to the safety limits. This advantage was properly taken into account when the simulation of refueling was conducted.

In this direction a preliminary study series was done to analyze to what extent can the existing ROP system for natural Uranium be used for SEU 0.9%. In ref. /14/, O. Nainer determined the critical parameters associated to a limited set of 48 distributions selected and computed by V. Raica in ref. /15/. Although they have an introductory character, these studies allow one to be aware of the possibility of using the detectors system associated ROP for natural Uranium, eventually considering a new set of starting thresholds.

5. The reactivity devices performance evaluation

One of the physics analyses objectives for SEU reactors is to demonstrate that the reactivity devices can perform the same functions as for the natural Uranium reactor. This means two main issues:

- setting the requirements, that is indentifying how much these must be modified, in terms of the reactivity need, at the change between natural Uranium and SEU;

- the devices efectiveness modification evaluation at the switch between natural Uranium and SEU.

The requirements modification appraisal was made based on the previous assessments of the reactivity effects.

The modification of the devices effectiveness has two major reasons:

- the devices perturbing effect modification that must be identified at super cell level and comes into the incremental cross sections modification which characterizes the particular device;

- the flux macrodistribution modification at the core level that implies the modification of the weight the respective perturbation has in the neutron balance. This effect is treated at the same time with the core flux and power distribution.

Following, a brief presentation of the principal results on these reactivity devices performances evaluations done by P. Laslau in ref. /16/ is shown.

The mechanical control absorbers:

The preliminary results indicate a reduction from 11 mk (for natural Uranium) at 9 mk. For SEU fuel at equilibrium, the positive reactivity that must be covered at reactor shutdown is less than 4 mk, having a temperature reactivity coefficient with 10 % less than that for natural Uranium. The value is greatly overcome by rods effectiveness. In reality, the rods were dimensioned to satisfy this requirement for fresh fuel, situation in which the temperature reactivity effect is more stressed but does not present interest for these analyses, in which one can assume the equilibrium comes by transition from natural Uranium.

The liquid zone control system:

The fuel burnup results in a reactivity variation that changes from 0.4 mk/d for natural Uranium to 0.5 mk/d for SEU. The changes in thermalhydraulic condition imply positive or negative reactivities with respect to the changing parameter and they are generally smaller for SEU with about 10 %. The reactivity need for power level modification depends on required speed and dynamic characteristics of the core. One can consider that all these needs will be generally smaller, taking into account that both the lifetime and the delayed neutron fraction have smaller values.

Preliminary evaluations indicated a total zone control system efficiency from 7 mk for natural Uranium to 5.5 mk for SEU. Till now, it was not identified any situation in which the zone control system could not properly react.This result may be explained either by the reduction of reactivity requirements at the switch between natural Uranium and SEU or by the possible overdimensioned system capacity for natural Uranium. A particular aspect which confirms this statement is linked with the control system capacity to suppress Xenon oscillations, that were analyzed in full detail together with the Xenon effects. In figure no. 9, which presents the left-right flux divergences evolution for setting off the Xenon divergent oscillations for the first azimuthal mode, the same deformations evolution in the case of control system activation is shown. One can see the difference is reduced in the magnitude and damps rapidly. This aspect stands out in figure no.11, where the front-back first axial mode induced deformation is shown.

(refueling simulations)

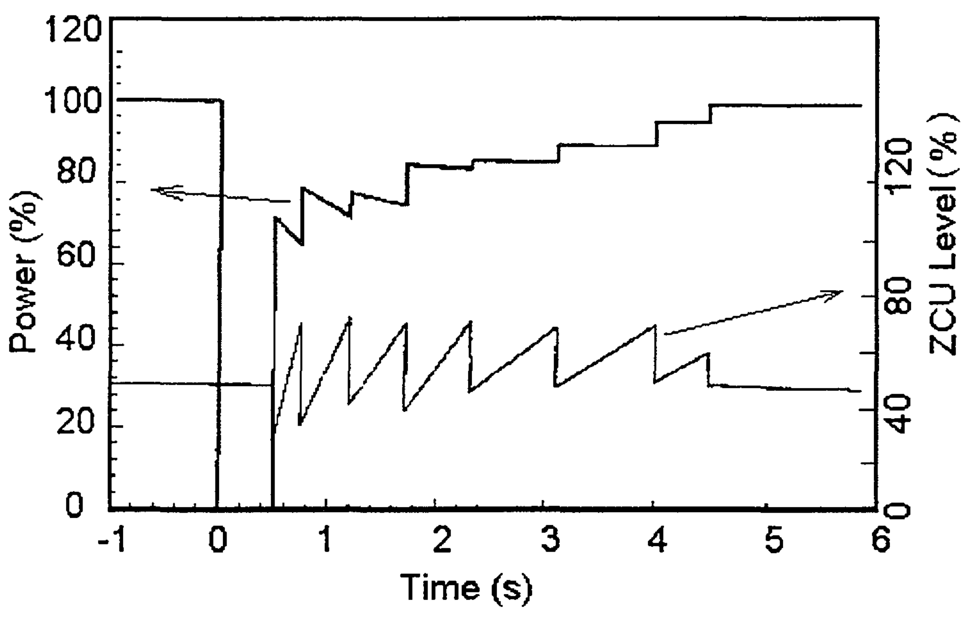

Figure 18. Restart power transient, actual scheme

The adjuster rod system

From the point of view of requirements, the value of reactivity margin is reduced both for control and for Xenon. Keeping in mind that the adjuster rods are permanently present in the core, influencing directly the power distribution, the adjuster rod system was treated in more detail, even in these preliminary evaluations. The adjuster rods role was virtually treated in the equilibrium core calculations by determining the power distribution that complies with the design requirements.

P. Laslau analyzed the possibility of restarting after a short shutdown, demonstrating in ref. /17/ that the adjuster rods can be used for this situation in the same manner as for natural Uranium, as one can see in figure 18, in which this dynamic analysis is shown. In ref. /18/ one analyzed the reactivity shim at power reductions. The calculations show that at a power reduction of 50 %, the reactivity due to Xenon reaches after 2.4 hours a maximum of 39.6 mK, compared with 27.8 mK initial value.

This represents the limit situation that could still be compensated by adjuster rods system, considering their reactivity is 11.8 mK, and by their extraction, the flux deformation decreases the shape factor to 0.356. In this paper, one analyzed the possibility of using the adjuster rods for compensating the negative reactivity induced by fuel burnup, in the situation of loss of refueling, determining a prolongation of operation by 2 days (at the same time with the power reduction up to 95 %) by the second bank extraction. These results were obtained for the actual scheme of banks, the analyses proving that for SEU fuel there are more effective grouping possibilities, regarding the functioning at refueling capacity loss and with similar restarting performance.

6. Shutdown system performances

For natural Uranium, the shutdown system performances are evaluated on the power transient basis, in the LOCA time. For that purpose, for SEU 0.9% one must analyze to what extent this transient evolution is modified by reactivity effect increment due to the void and by the shutdown system effectiveness reduction. An additional problem is associated with the uncertainties related to the prediction of the void effect reactivity for the irradiated fuel case. The differences between WIMS and POWDERPUFS estimations are well known even for natural Uranium.

To illustrate this, in figure no. 19 the power evolution for natural Uranium fed core is shown, computed by WIMS and PPV constants for the same evolution of coolant density (corresponding to a RIH break of 20 %). The evolution is more rapid for the transient computed by WIMS constants, reaching a maximum of 160% of the nominal power, compared with 140 % obtained in the PPV based calculations. This is explained by the more emphasized void effect predicted by WIMS.

For the SEU 0.9 % reactor, one has a higher value of void effect for the fresh zone. For equilibrium fuel yet, one has some small differences between natural Uranium and SEU for void effects. This is the result of the void effect behavior which diminishes with burnup following the Pu gathering. The void effect for SEU, though

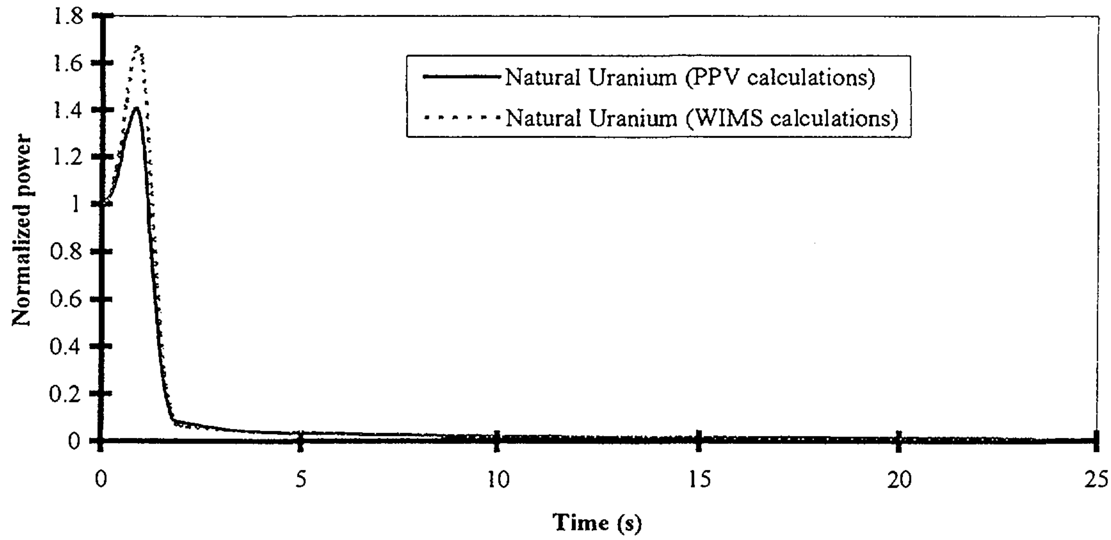

Figure 19. Power transients for LOCA (Natural Uranium)

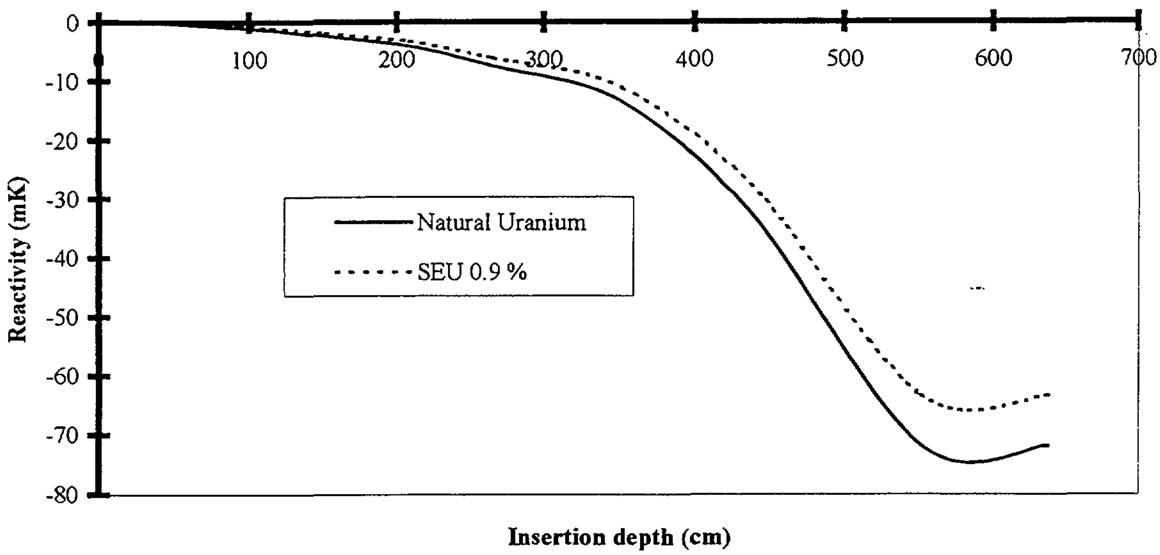

Figure 20. Reactivity insertion for Natural Uranium compared with SEU 0.9 %

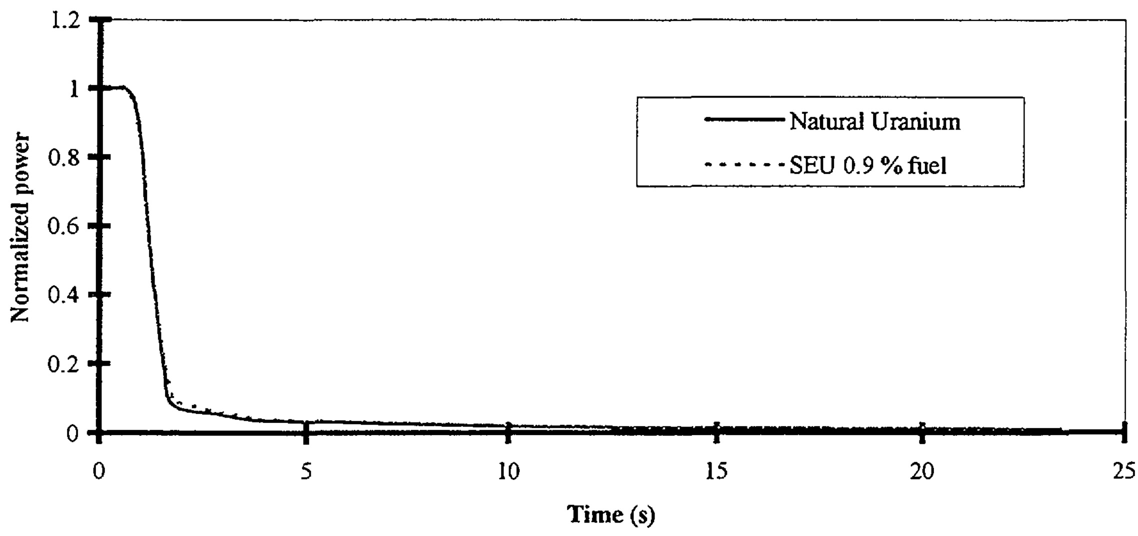

Figure 21. Power transients for reactor shutdown (SDS 1)

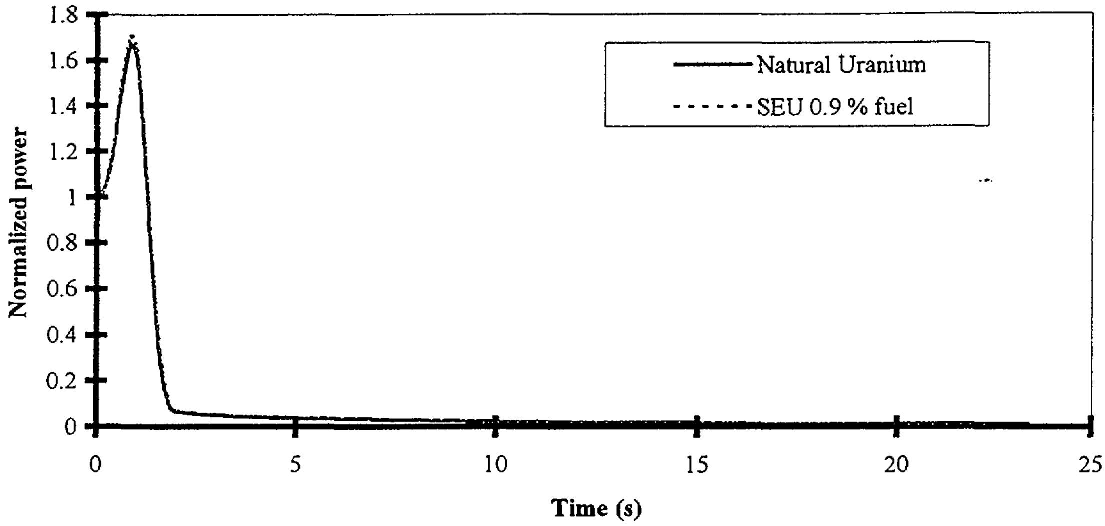

Figure 22. Power transients for LOCA

initially has a higher value, will present a sharper reduction because of the almost double discharge burnup. In this way, for SEU 0.9 % at equilibrium, one will obtain values close to those obtained for natural Uranium, effect predicted both by PPV and WIMS.

For the SDS 1 efficiency modification, compared with natural Uranium, for SEU 0.9 %, though one can see a reactivity insertion curve difference illustrated in figure no. 20, the effect in power evolution is greatly diminished, especially on the first part where usually the critical conditions are reached, as one can see in figure no. 21 where the power transients generated by rods insertion for the two types of fuel are shown.

A comparison of power evolution for the two types of fuel is shown in figure no. 22. This shows an assessment of the shutdown system capacity to limit the power evolution in the LOCA case, more exactly an assessment of the modification of this capacity at the switch between natural Uranium and SEU 0.9 %. To set off the differences, WIMS calculations were done both for natural Uranium and SEU 0.9 %, considering the same density evolution curve. One can remark the very close evolutions, the differences being of a few percent order, containing both the void effect modification and the efficiency reduction at the switch between natural Uranium and SEU fuel.

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