Study of the nuclear fuel behavior with coupled three-dimensional neutronics/thermal-hydraulic codes

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Fuel behavior of a WWER-1000 reactor during a Rod Ejection Accident (REA) is studied in this paper. Tools used are a coupled three-dimensional (3D) neutronics / thermal-hydraulic codes for the study of the transient and a 2D neutronics transport code for the generation of the cross sections libraries. Sensitivity analyses were performed varying various physical parameters. The effect of different cross section modelling and the effect of various states of operation of the plant on the transient evolution were also investigated.

KEYWORDS: Nuclear Fuel Behavior, Energy Release to the Fuel, Rod Ejection Accident, Coupled Codes, Cross Section Generation

1. Introduction

Increasing the burnup of the nuclear fuel is one of the requirements for an economic performance of a nuclear power plant and therefore a general trend in the nuclear industry. However, this tendency implies that the fuel has to be exposed for a longer time in a severe ambient. The last experiments performed suggest that increasing the burnup could lead to a stricter limit for the energy released to the fuel during the Reactivity Insertion Accidents (RIA).

In this contest, the safety analysis of the WWER-1000 core, assumes great relevance. These types of reactors, in fact, have just started to improve their burnup following the trend of the industry in the Western countries.

The aim of this work is to perform a safety analysis of the nuclear fuel of a WWER-1000 during the control rod ejection accident using a three dimensional (3D) neutronics code coupled to a thermal-hydraulic code for the transient analysis. A 2D neutronics transport code for the generation of the cross section libraries is utilized. The analysis was performed on the fuel of a core at the Beginning of Life (BOL) and at the End of Cycle (EOC). The term BOL is used in this paper to refer to beginning-of-cycle (BOC) of first cycle. In this way, it was established a methodology to execute further analysis in the future on fuel with an increased burn-up.

The work was carried on by a collaboration between the Pennsylvania State University and the Università di Pisa. Particular care was dedicated to the comprehension of the effects of different cross section libraries on this transient and on the neutron kinetics code in general. Moreover sensitivity analyses were also performed varying the physical properties of the fuel and the flow regime of the plant, in order to understand the effects of a single parameter variation on the trend of the transient.

This safety analysis was performed following the IAEA guidelines for accidents analysis of WWER NPPs [1] and accounting for the results of the main experiments on the RIA [2],[3].

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2. Description of the work

2.1 Reactor Description

This study used as reference Nuclear Power Plant (NPP) the Kozloduy-6 NPP, a WWER-1000, model 320 [4]. It is a pressurized water reactor with a thermal nominal power ($P_{Nominal}$) of 3000 MW, equipped with 4 loop and 4 horizontal steam generators. The reactor core is composed by 163 hexagonal fuel assemblies with ^{235}U enrichment between 2.0 and 3.3 %. Every fuel assembly is composed by 330 fuel pins and a water rod. Sixty-one of these fuel assemblies have 18 guide tubes to allow the insertion of the control rods.

2.2 Accident Description

The control rod ejection accident is one of the Design Basis Accident (DBA). It is assumed a failure of control rod mechanism pressure housing such that the reactor coolant pressure could eject the control rod to the fully withdrawn position. Consequences of this event are a rapid insertion of reactivity and therefore a power burst, with great increase of the fuel temperature. At the same time this could cause fragmentation of the fuel, a DNB on the clad surface and a rupture of cladding with consequently fuel dispersal into the coolant.

The transient is terminated by the Doppler effect, by the negative moderator temperature coefficient and eventually by a reactor scram. It is also assumed that the break in the pressure housing is closed by the ejected control rod, preventing the consequent small-break loss-of-coolant-accident (SBLOCA) .

3. Modelling and tools used for the analysis

The 3D neutron kinetics code PARCS [5] was used for the transient analysis coupled with the thermal-hydraulic code RELAP5 [6]. The codes' spatial coupling is executed through a special file, the MAPTAB file, which allows to couple the fuel assemblies with the right thermal-hydraulic channels.

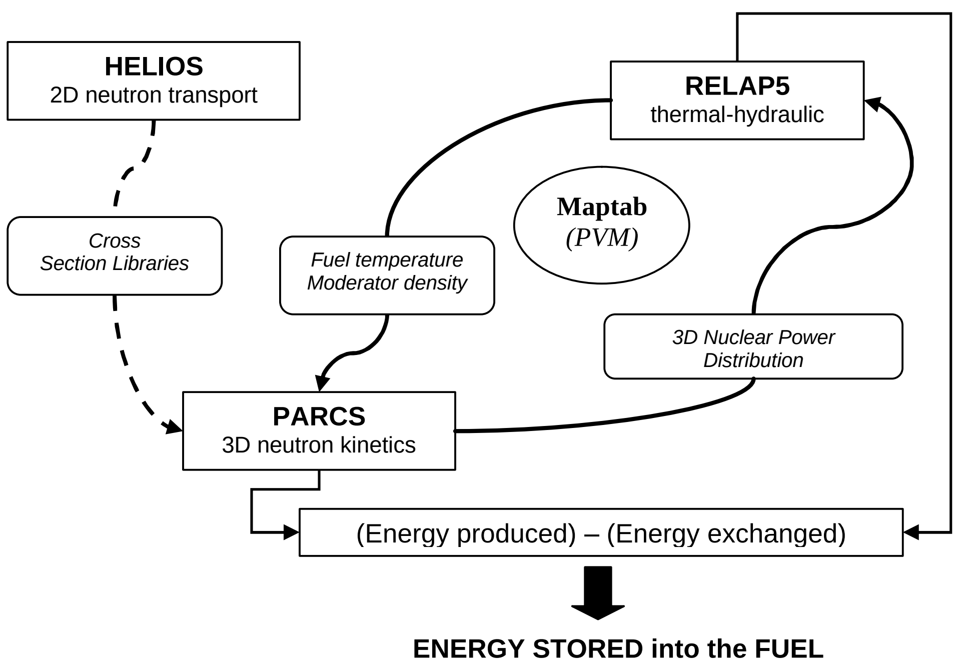

Fig. 1 Calculation Flow

The coupling software is based on the Parallel Virtual Machine (PVM). The cross sections used by PARCS are computed in a separate process by the 2D neutron transport code HELIOS [7].

3.1 Neutron Cross Section Modelling

Cross section libraries were developed using the 2D lattice physics code HELIOS. Detailed input files were created to describe the hexagonal fuel assemblies of the core.

Twenty-eight different types of fuel assemblies were identified for the problem's modelling and each of these were divided axially in 10 planes in order to reconstruct the different burnup of the fuel along its height. All data of the burn-up of the fuel were obtained from the Kozloduy 6 NPP.

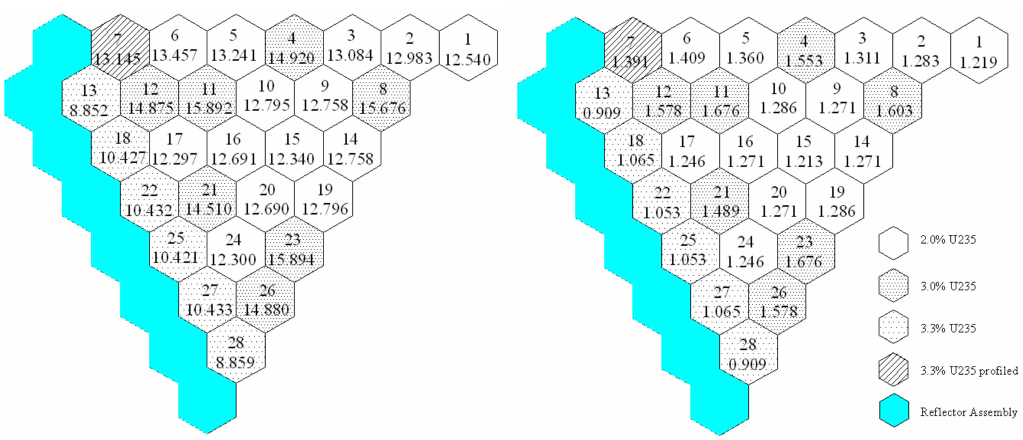

Fig. 2 1/6 Core Lay-out at EOC and BOL (burnup expressed in MWd/KgU)

Therefore 280 composition cross-section set calculations for the un-rodded fuel assemblies plus 110 for the rodded fuel assemblies are needed to obtain a complete cross section library. Six different cross section libraries were developed; their different physical characteristics are described in Table 1 and briefly summarized below:

Table 1 Characteristics of calculations for the cross section libraries

Library's name	Average Burnup (MWd/KgU)	Fuel Temperature Range (°K)	Moderator Density Range (kg/m^3)
Xsec_20_BOC	1	540 to 1500	786 to 661
Xsec_20_EOC	14	540 to 1500	786 to 661
Xsec_42_EOC	14	540 to 2000	786 to 103
Xsec_20_Xenon	14	540 to 1500	786 to 661
Xsec_20_Beta	14	540 to 1500	786 to 661
Xsec_20_Gap_closure	14	540 to 1500	786 to 661

- Xsec_20_BOC is modelling the nuclear fuel at the BOL
- Xsec_20_EOC is modelling the nuclear fuel at the EOC, with a medium burn-up
- Xsec_42_EOC was created in order to understand the effect of an increased number of reference points (from 20 to 42)

- Xsec_20_EOC_Xenon is modelling the Xenon poisoning of the core at its maximum peak (roughly 9 hours after a scram); the fuel assembly where the rod ejection happens is not poisoned because it is assumed that the control rod was inserted a lot of the time before the scram that caused the Xenon poisoning
- Xsec_20_EOC_GapClosure is modelling the gap closure and the change in thermal conductivity for the fuel assembly where there will be a rod ejection
- Xsec_20_EOC_Beta allows calculations with the use of 'local' delayed neutron fractions for each of the 280 fuel compositions, instead of the use of a core-wide averaged values

3.2 Thermal-hydraulic modelling

The RELAP5 code was used for the thermal-hydraulic calculations. A complete nodalization was developed, describing all the primary side and the secondary side of the plant until the turbine.

Particular care was dedicated to the description of the core, where 41 thermal-hydraulic channels were modelled. Fourteen of these are channels which model only one fuel assembly. In this way it was possible to study also the behavior of the fuel assemblies near the assembly that experienced the rod ejection. Every channel is divided axially into 20 layers in order to obtain a good axial resolution (0.1775 m).

3.3 3D neutron kinetics modelling

3D neutron kinetics code PARCS was used for the nuclear calculations during the transient studied. All the reactor core, composed by 163 assemblies with a pitch of 0.236 m, was modelled. Thanks to the symmetries, there were utilized 28 or 29 different fuel assembly types. Every fuel assembly was divided axially into 20 layers; moreover, bottom and upper axial reflectors were considered. Further 48 assemblies, on the edges of the core, simulate the radial reflector.

The 61 control rods, grouped in 10 banks (I to X), are also simulated. A total of 4642 neutronic nodes were finally used for the calculations.

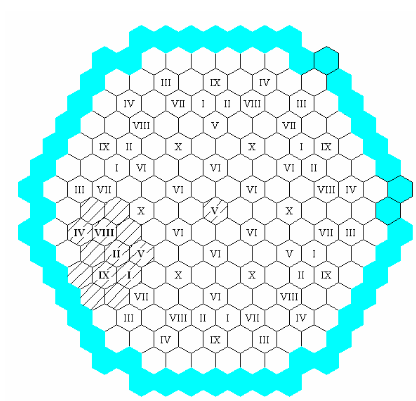

Fig. 3 Core Lay-out with the CRs banks locations, the reflector assemblies and, marked with an hatch, the 14 thermal-hydraulics channels, which are coupled to single neutronics assembly

3.4 Developed tools

Special control variables were created in the RELAP5 in order to calculate the energy exchanged by the fuel assemblies with the coolant. Thanks to an 'ad-hoc' Fortran program it was also possible to calculate the energy production in every fuel assembly. An other interpolating program allowed to estimate the energy stored into the fuel utilizing the data from the previous control variables and from the Fortran program**.**

4. Transient simulations

The Hot Zero Power (HZP) and Hot Full Power (HFP) with the nuclear fuel at the BOL and at the EOC were analyzed. Previous calculations were executed in order to estimate the control rod with the maximum worth for each of these four states. The control rod was always assumed to be ejected in 0.1 seconds from the fully inserted to the completely withdrawn position. The scram delay signal was assumed to be 0.3 seconds and the time for the complete insertion of all the CRs was assumed to be 3.6 seconds. In table 2 are summarized the main parameters imposed for the defining of the plant status.

Table 2 Main plant parameters for the HZP and the HFP

	Hot Zero Power	Hot Full Power
Core Power	0.1 % $P_{Nominal}$	100 % $P_{Nominal}$
PRZ pressure	15.6 MPa	15.6 MPa
Pumps in operation	4 / 2 / 1 / 0	4
Boron Concentration (ppm)	1208 (BOC) 5.0 (EOC)	1208 (BOC) 5.0 (EOC)
Core inlet coolant temperature (°K)	550	550
Fuel average temperature (°K)	552	977
CRs gruop positions	I to III A.R.O.(*) IV 57% withdrawn V A.R.O. VI to X A.R.I.(**)	I to X A.R.O.
Transient	1 CR of IX bank ejected in 0.1 seconds Scram set-point : 40% $P_{Nominal}$	1 CR of the II bank ejected in 0.1 seconds Scram set-point : 120% $P_{Nominal}$

(*) All Rods Out
(**) All Rods In

4.1 Sensitivity analyses

The HZP-EOC case was chosen among the others for sensitivity analyses because it was the case that maximize the energy released. It was analyzed, varying the flow regimes of the plant turning off all, three and two of the four Main Coolant Pump, as suggested by the IAEA guidelines [1].

It was also studied :

- the effect of the Doppler weighting factor variation
- the effect of changing physical properties of the fuel (i.e. geometrical dimension, density)
- the effect of Xenon poisoning
- the effect of cross section libraries with an increased number of reference points (from 20 to 42)
- the effect of cross section libraries with delayed neutron fractions calculated for each fuel type

4.2 Execution of transients and data processing

Thanks to the tools developed (see 3.4) it was possible to calculate the radially averaged fuel enthalpy for every layer of the fuel assemblies, identifying the location of the maximum value. It was assumed as a threshold value for the cladding failure, the value of 160 cal/g, considering the low burn-up of the fuel and the studies executed on the WWER clad during the RIAs [2].

It was also possible to identify the hot spot and the values, in this location, for the clad and fuel temperature trends during the transient.

5. Results

Table 3 and Figure 4 summarize the results of the calculations showing the maximum enthalpy stored into the fuel during the various simulated transients.

Table 3 Transient's results for the maximum energy released to the fuel

Reactor & Fuel State	Cross Section Library	Maximum Fuel Enthalpy (cal/g)
HZP-BOC	XSec_20_BOC	26.2
HZP-EOC	Xsec_20_EOC	81.0
HFP-BOC	Xsec_20_BOC	63.9
HFP-EOC	Xsec_20_EOC	59.5

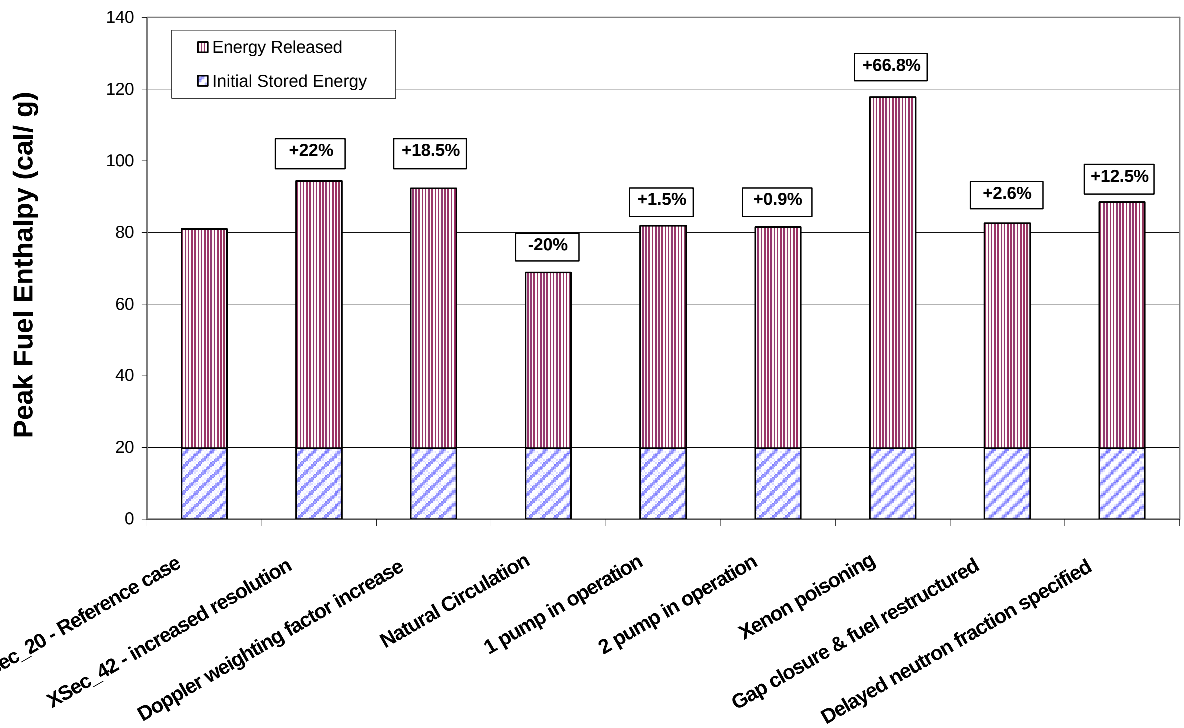

Fig. 4 HZP EOC sensitivity analyses – Peak Fuel Enthalpy

Regarding these values it could be noted that:

- HZP EOC is the most severe transient for the fuel, compared to the other reference transients (HFP BOL, HFP EOC and HZP BOL)

- Cross section library with an increased number of reference points causes a modest increase (+21.9%) of the energy deposited; the greater number of reference points means a better interpolation and, in this case, it resulted in a decrease of the Doppler effect.
- The variation of the flow regime of the plant decreases the energy deposited only if there is natural circulation established. This reduction is due mainly to the worst heat transfer and consequently to its influence on the effect of the negative moderator and fuel temperature coefficients. On the other hand, there is a greater increase in the clad maximum temperatures (+31% compared to the clad maximum temperature of the reference case)
- Xenon poisoning of the core caused the maximum energy deposited (+66.8%) compared to the reference case. It was explained by an increase of the control rod worth due to the greater flux redistribution from the top to the bottom of the core
- Gap closure and fuel restructuring do not affect the value of the maximum energy deposited but they affect the clad maximum temperature (+55% increase)
- Variation of Doppler weighting factor from 0.1 to 1.0 increases modestly (+18.5%) the energy deposited.
- Cross section library with delayed neutron fraction (beta) values specified for every fuel type instead of a single core wide averaged value, increased the energy released (+12.5%). This was explained by a neutron flux redistribution in zones of the core with beta values smaller than the core averaged beta value

The location of the maximum energy released was always identified for the HZP cases in the fuel assembly that experienced the rod ejection, generally at height of 3.2 m measured from the bottom part of the active core.

Thereafter, in order to maximize the energy released, the temperature of the fuel and the temperature of the clad, it was run a transient combining:

- the effect of the Xenon poisoning
- the increase of the Doppler weighting factor
- the reduction of flow regime, turning off 2 of 4 main coolant pump

The maximum energy released to the fuel was roughly 118 cal/g, smaller than the 160 cal/g constraint for WWER-1000 clad failure. The fuel and the clad maximum temperature were not of safety concern, but a massive ebullition of the water and DNBR lower than 1.0 was found in the thermal-hydraulic channel. Cautiously, according to the IAEA guidelines [1], it would have to assume that the fuel rods fail. The trend of the main parameters for the worst case scenario are shown in the figures below.

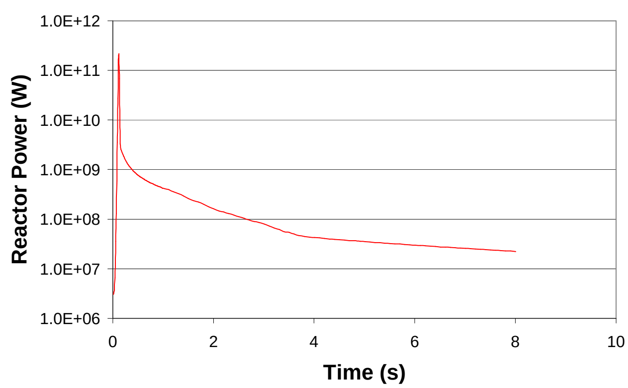

Fig. 5 Reactor Power

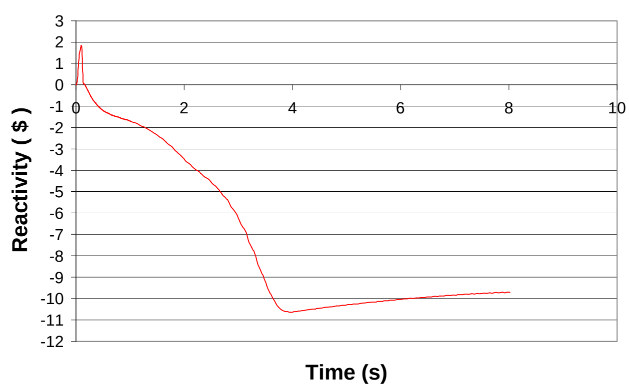

Fig. 6 Reactivity trend

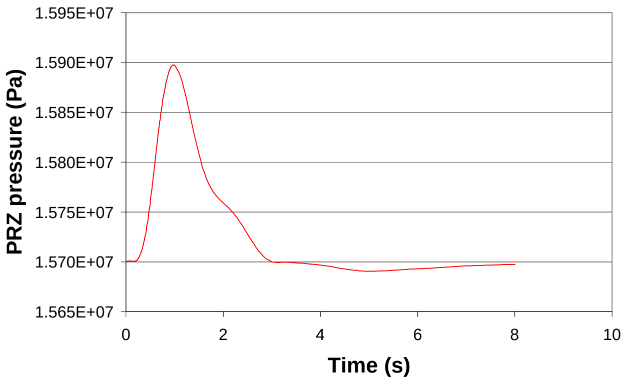

Fig. 7 Pressurizer pressure

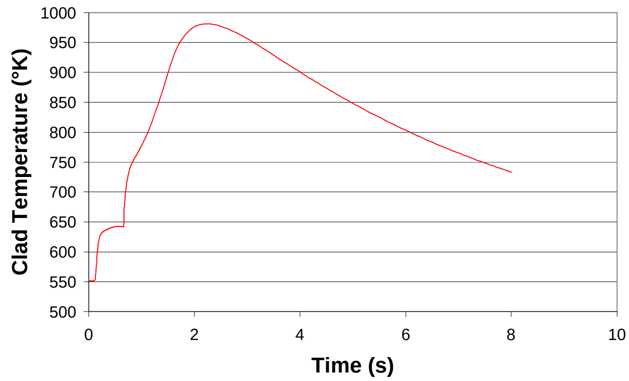

Fig. 8 Clad temperature at the Hot Spot

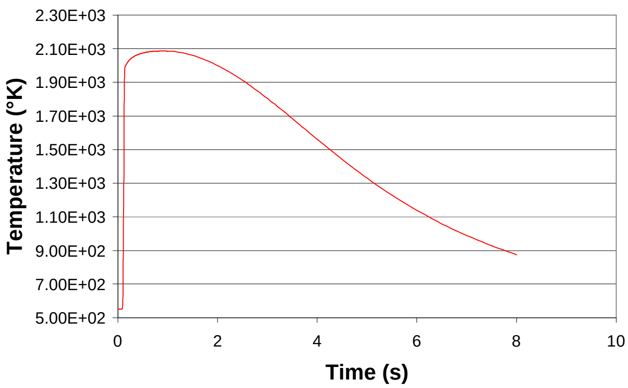

Fig. 9 Fuel Center-line temperature

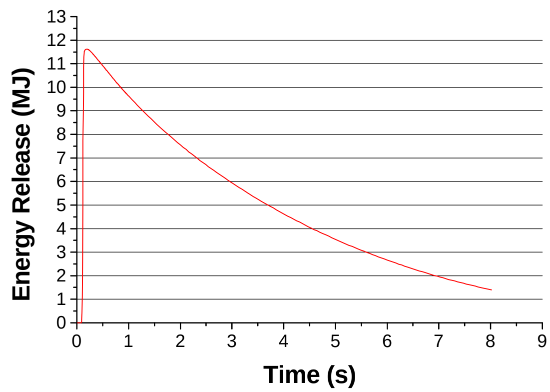

Fig.10 Maximum Energy Released at the Hot Spot

6. Conclusions

Safety analysis of the WWER-1000 core during a REA was executed. HZP and HFP conditions for the BOL and EOC core were analyzed; sensitivity analyses were also carried on using different cross section libraries and varying a wide spectrum of physical parameters of the fuel and of the plant.

It was found that during the worst possible scenario (HZP-EOC, only 2 pump in operation, Xenon transient for the core) the maximum energy deposited to the fuel was below the failure threshold value generally adopted for the WWER clad, but it was also found that the fuel assembly experienced a DNB.

In conclusion, further studies are needed, especially because the current safety limits could be changed in future for high burnup fuel to much lower value.

References

1. “Guidelines for accident analysis of WWER Nuclear Power Plant”, IAEA, vol. 28, December 1995
2. L. Yegorova, F. Schmitz, J. Papin “Mechanical Behavior of Fuel Element during RIA Transients”, Proc. of the Topical Meeting “Physics, Safety, and Applications of Pulse Reactors”, Washington DC, USA, November 13-17, 1994
3. M. Champ, J. Lewi, JP Merle, J. Papin “Behavior of PWR’s Fuel under Reactivity Insertion Accident (RIA)” Proc. of 5th Int. Conf. on Nuclear Engineering, ICON 5, Nice, France, May 26-30, 1997, ICONE - 2467

4. Ivanov, B.. et al., “WWER-1000 Coolant Transient Benchmark-Phase 1”, NEA/NSC/DOC (2002) 6
5. PARCS 2.43 code manuals, Purdue University
6. “HELIOS Methods”, Studsvik ScandPower, (2000)
7. RELAP5/MOD3.3 code manuals, INEEL