PROGRESS IN THE ANALYSES OF IRT, SOFIA LEU CORE: ACCIDENT ANALYSES

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ABSTRACT

The initial LEU (IRT-4M fuel assemblies, 19.75% ^{235}U) core of the new IRT, Sofia research reactor of the Institute for Nuclear Research and Nuclear Energy (INRNE) of the Bulgarian Academy of Science, Sofia, Bulgaria is jointly analyzed with the RERTR Program at Argonne National Laboratory (ANL) to evaluate its response to transients/accidents. The initial configuration using 16 fuel assemblies (four 8-tube and twelve 6-tube fuel assemblies) is analyzed at a critical core state preferable for BNCT tube operation. Results of accident analyses performed for this critical state are presented in this paper and will be included in the IRT, Sofia Safety Analyses Report. These results show that safety is maintained for all transients analyzed.

1. Introduction

A joint study concerning IRT, Sofia research reactor (RR) between INRNE and the RERTR Program at ANL was initiated in 2002. The previous steps [1-4] were mainly focused on neutronics properties significant for reactor application and safety analyses. Presented here are results of further analyses significant for safety assessment of the LEU core for the critical state preferable for Boron Neutron Capture Therapy (BNCT) beam tube operation. The BNCT activity is considered nowadays as one of the most important for future IRT, Sofia application. That is why accident analyses are performed using this critical state as the base one. Results of neutronics calculations that determine the IRT, Sofia Safety System properties and its behavior during accidences are presented in a supplementary paper [5]

The following accident scenarios have been analyzed to assess the response of the reactor to the specified conditions, and specifically to determine if any fuel damage occurs as a result of the accident:

1. Insertion of reactivity by uncontrolled withdrawal of a control rod from its critical position.
2. Insertion of reactivity by disengagement of the aluminum follower of one control rod.
3. Insertion of reactivity by ejection of a movable experiment.
4. Insertion of reactivity by flooding of the air gap of one water "displacer" core element.
5. Insertion of reactivity by cold water injection into the inlet of the core.
6. Loss of forced flow (LOF) accident because pumps stop as a result of loss of the offsite electricity supply.
7. Blockage of inner channel of fuel assembly.
8. Unbalanced control rods positioning.
9. Core loading accident.
10. Blockage of fuel assembly: Beyond Design Basis Accident.

Transient analyses are used for description of the first six above mentioned accidents. Steady-state thermal-hydraulic calculation is sufficient for description of the seventh accident. The analyses of accidents 8 and 9 above are based on the neutronics calculation [5]. The tenth accident in which one fuel assembly is assumed to melt and release of radioactivity occurs is defined as the Beyond Design Basis Accident (BDBA).

2. Reactivity Insertion Accidents

The analyses of the reactivity induced transients were performed using the PARET (v7.3) code [6]. The PARET code uses a point kinetics model to calculate the total reactor power as a function of time after the transient initiation. Thermal-hydraulic feedbacks to the core reactivity due to fuel and coolant temperature changes, and coolant voiding are modeled using reactivity feedback coefficients. The transients' reactivity values and feedback coefficients were calculated and are presented in reference 5. Multiple fuel-coolant channels modeling ability is available in PARET. Multiple channels are used for a self-consistent description of different portions of the same core. In this analysis, the hottest fuel tube was modeled, along with the rest of the core modeled as the average channel; i.e. two channels were used in the PARET model.

For all reactivity induced transients it was assumed that the reactor is normally operated at 1000 kW power and with one pump in the primary cooling circuit. Also, the coolant inlet temperature was assumed to be 45ºC for all reactivity transients, with exception to the cold water insertion into the core.

The reactor, at power, has a trip signal on period (less than 10 s) and on power level (20% above nominal operating power) [7]. In all the analyses below it is conservatively assumed that the period trip does not work even if the set point is reached, and that the reactor will scram on power level. Note that the period trip is not reached for essentially all the transients analyzed below. After scram, all safety, shim, and automatic rods are inserted into the core after a signal delay of 200 ms and with an insertion time conservatively assumed to be 800 ms. Here, it is also conservatively assumed, in the analyses below, that only the safety rods participate in the scram, and the respective reactivity insertion trace is calculated using safety rods cumulative worth dependence on the depth of insertion presented in the supplementary paper [5].

2.1. Uncontrolled withdrawal of a control rod from its critical position

Two transients, withdrawal of shim rod KO-1 (highest worth rod) and automatic rod AR are analyzed. Both transients are assumed to be initiated by the upwards movement of the corresponding rod from its critical position to the fully withdrawn position. The uncontrolled withdrawal speed of both rods is assumed to be the maximum design withdrawal velocity [7], i.e. 8 mm/s for the shim rod and 50 mm/s for the automatic rod. The respective reactivity insertion traces are calculated using KO-1 and AR rods worth dependence on the depth of insertion presented in [5]. The total reactivity insertion for KO-1 rod is equal to 2.01 β and 0.22 β for AR rod.

The results for the clad surface/fuel/coolant temperatures for the hottest fuel tube and channel are illustrated in Figures 1, 2.

These results show that the maximum temperature on the clad surface during KO-1 transient is 87°C and 88°C for AR transient. Both temperatures are below the temperature for onset of nucleate boiling (115°C) [5] and well below the clad softening temperature limit of 425°C [8].
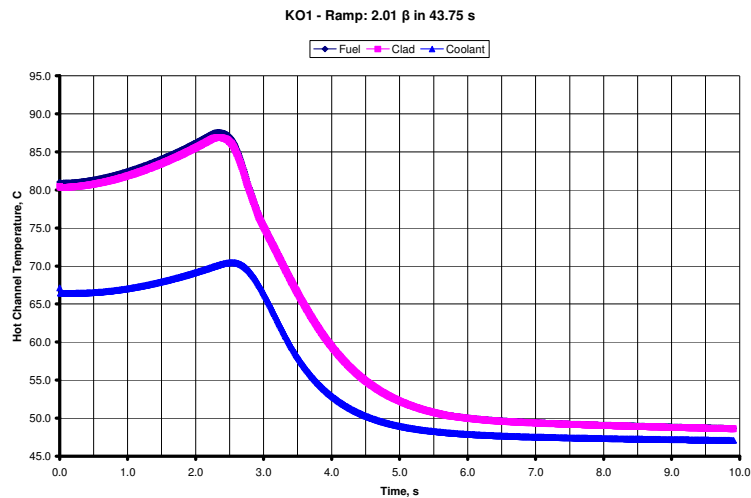

Figure 1 Uncontrolled Withdrawal of Shim Rod KO-1: Clad/fuel/coolant temperature
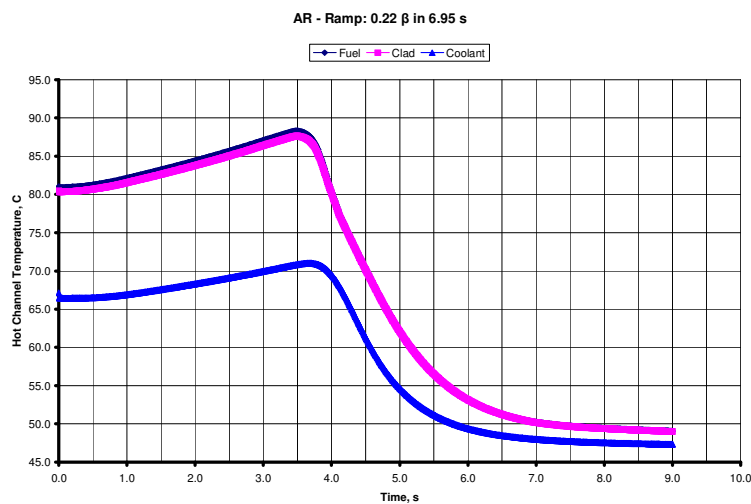

Figure 2 Uncontrolled Withdrawal of Automatic Rod AR: Clad/fuel/coolant temperature

2.2. Disengagement of the aluminum follower of a Control/Safety Rod

Accidents with the followers of the safety rod AZ-1 and shim rod KO-1 are considered. Both transients are assumed to be initiated by the disengagement of the aluminum follower of the corresponding rod and falling down from its critical position. The safety rod AZ-1 is fully withdrawn during operation Its follower disengagement was modeled as a reactivity insertion of 0.35β (total worth of the follower) in 0.8 s, and the respective reactivity insertion trace is calculated using AZ-1 rod follower' worth dependence on the distance from its critical position presented in [5]; the time for the fall of the follower was assumed to be the same as for the insertion of the rods after scram. The shim rod KO-1 follower disengagement was modeled as a reactivity insertion of 0.32 β in 0.58 s; the respective reactivity insertion trace is calculated using the results presented in reference 5. The time for insertion was calculated using the same acceleration as that for the insertion of the safety rods after scram.

Both transients' results for the clad surface/fuel/coolant temperature for the hottest fuel tube and channel are shown in Figures 3 and 4.
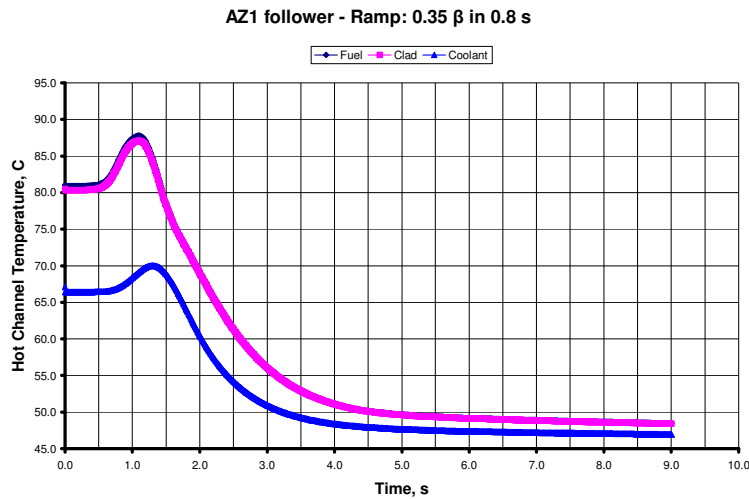

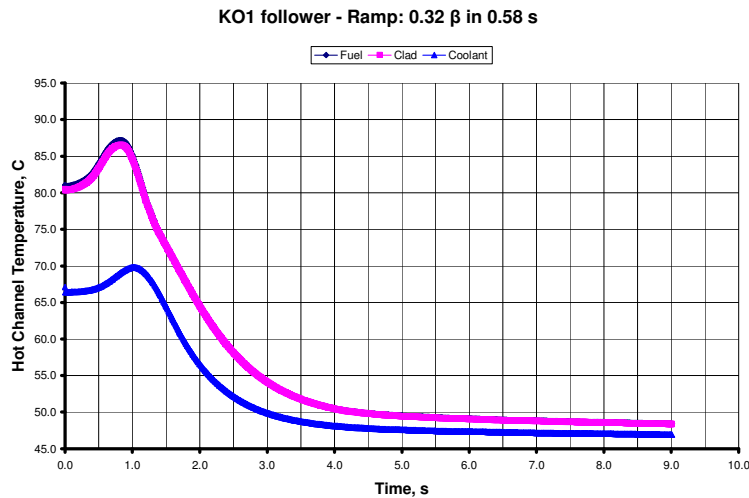Figure 3. AZ-1 Follower Disengagement: Clad/fuel/coolant Temperature

Figure 4. KO-1 Followers' Disengagement: Clad/fuel/coolant Temperature

The results above show that the maximum temperature on the clad surface is 87°C during both transients. This temperature is below the temperature for onset of nucleate boiling (115°C) and well below the clad softening temperature limit of 425°C.

2.3. Reactivity Insertion due to Movable Experiment

In this transient it is assumed that the maximum reactivity worth of a movable experiment is equal to 0.5β. It is further assumed that this reactivity is practically instantaneously (0.05 s) inserted into the core; i.e. almost a step experiment ejection.

The results of the PARET analysis for the reactor power level and the clad surface/fuel/coolant temperature for the hottest fuel tube and channel during transient are shown on Figures 5-6.
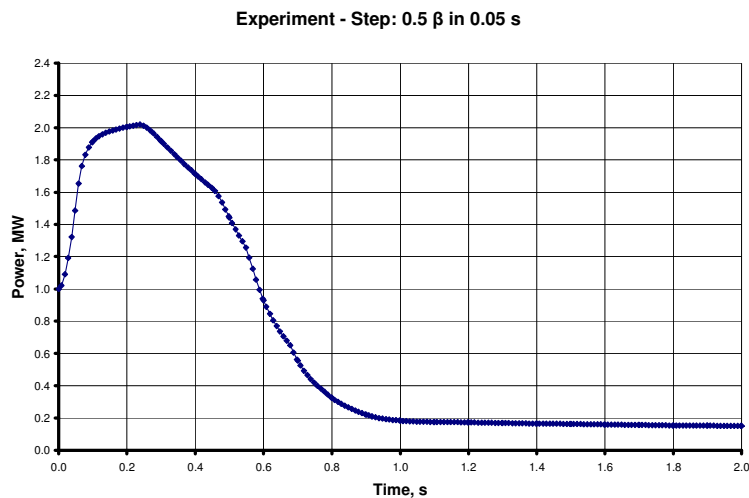
Experiment - Step: 0.5 β in 0.05 s

Figure 5. Ejection of Movable Experiment: Reactor power

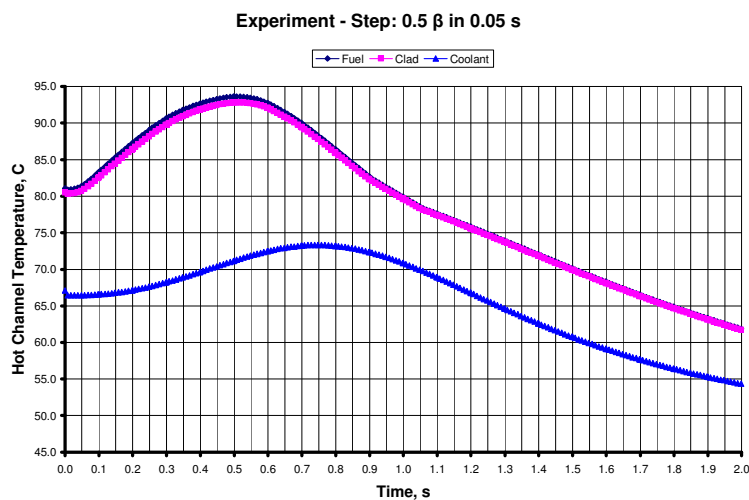**Experiment - Step: 0.5 β in 0.05 s**

Figure 6. Ejection of Movable Experiment: Clad/fuel/coolant Temperature

These results show that the maximum temperature on the clad surface during this transient is 93°C; this temperature is below the temperature for onset of nucleate boiling (115°C) and well below the clad softening temperature limit of 425°C.

The results obtained for the accident with movable experiment can be used for assessment of consequences of other accidents with rapid reactivity insertion. That is the case for the flooding of the air gap of one of the water "displacer" core elements in the reactor vessel (cells (H:I)(1:6)) [5] which results in an added reactivity equal to 0.22β and when it is assumed that this reactivity is inserted almost instantaneously in to the core; i.e. during 0.05s. Similar situation is for cold water insertion to the core. In this case it assumed that the reactor pool coolant instantaneous temperature drops from 20°C to 10°C. Based on the coolant temperature coefficient [5] this decrease in temperature would cause a reactivity increase of 0.13β. Both transients are similar to the reactivity insertion due to movable experiment but with a smaller reactivity value. That is why these transients will have a peak clad temperatures that are even smaller than that in Figure 6 above. It should be noted that the safety margin criteria is met for the reactor pool coolant temperature equal to 10°C. The shutdown margin is equal to - 1.24 %Δk/k.

3. Loss of forced flow (LOF) accident.

The reactor is operating at 1000 kW with one pump in the primary coolant circuit. When electric power is lost the reactor the control system is activated with delay less than 200 ms and the control rods drop to the core after 800 ms. According to calculations [9] the downward coolant flow coasts down over a period of 6 sec (Figure 7).
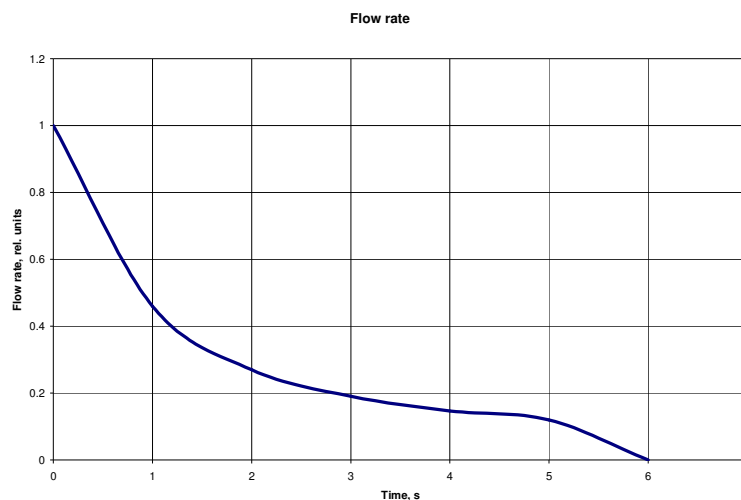

Figure 7. Downward coolant flow cost down

After the forced downward flow of the coolant stops the flow reverses to the natural convection mode.

The results of the PARET analysis are shown in Figures 8-9. Figure 8 shows the clad surface/fuel/coolant temperature for the hottest fuel tube and channel, and Figure 9 shows the coolant flow rate during the transient.

These results show that the maximum temperature on the clad surface during this transient is 81°C; this temperature is below the temperature for onset of nucleate boiling (115°C) and well below the clad softening temperature limit of 425°C.
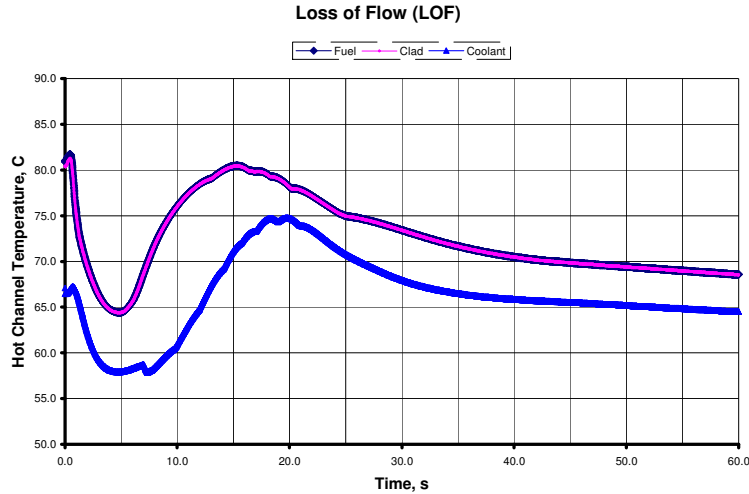
Loss of Flow (LOF)

Figure 8 Loss of Flow: Clad surface/fuel/coolant Temperature

Loss of Flow (LOF)
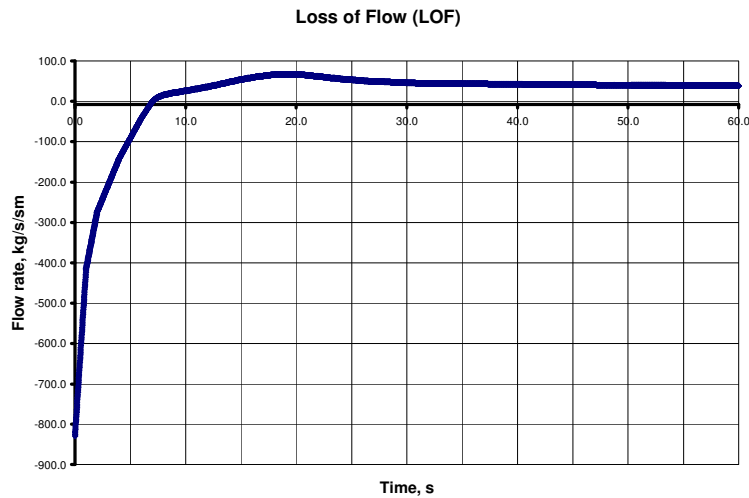

Figure 9 Loss of Flow: Flow rate

4. Blockage of inner channel of a fuel assembly

In this accident it is assumed that the central inner coolant channel of the hottest fuel assembly is blocked when reactor is operated with one pump and at 1000 kW power level. This accident will not be detected during operation. The reactor will continue to operate at the steady state condition at the nominal power.

The steady-state PLTEMP (v3.4) [10] code was used for evaluations of the consequences of this event. The calculations were performed accounting for the hot channel factors. The results of the calculation show that the maximum clad surface temperature in the blocked channel increases to 101.5°C (inner tube) The clad surface and coolant temperatures in the other tubes/channels is essentially not changed; there is only an increase of about 1°C in the tube closest to the inner tube. The minimum ONBR without the blockage (base case) is about 1.4 and it occurs in the second channel (between outer fuel tube and the next inner tube). In the case of the blockage of the inner channel the minimum ONBR decreases to 1.35 and its position is shifted to the channel that is the closest to the blocked one. This result show that

the maximum clad surface temperature is below the temperature for onset of nucleate boiling (115°C) and well below the clad softening temperature limit of 425°C.

5. Unbalanced control rods positioning and core loading accident

The influence of unbalanced control rods positioning on power peaking factors is evaluated in [5]. A set of five additional possible critical core states provided by different asymmetrical shim rods positioning are analyzed and compared. It is demonstrated that for this set of configurations the peak power varies only slightly; a maximum difference of less than 3% was calculated. The maximum power peak is equal to 88.0 $W.cm^3$ compared to 86.1 $W.cm^3$ for the base case analysis. Taking into account that for all slow enough accidents where the clad temperature is well correlated with the power level the maximum clad surface temperature is far below the limiting value it is concluded that the unbalanced positioning of the control rods will have a negligible impact in the safety of the IRT core configuration.

During the core loading, it is possible (assuming failures to follow loading procedure correctly) to load an 8-tube fuel assembly into the cell C3 instead of cell A3; because of this error, a 6-tube assembly is loaded in position A3. The influence of wrong loading of the fuel assemblies in the IRT core configuration on power peaking factors is evaluated in reference 5, where it is shown that the maximum power peak is essentially the same (in the limits of calculation uncertainty) as that for base case configuration analyzed. This means that even if a loading error occurs (and is not detected) it will have essentially no impact on the safety of the IRT core configuration.

6. Blockage of a fuel assembly. Beyond Design Base Accident

As the Beyond Design Basis Accident for the IRT-Sofia, it is postulated that by some means the flow to one fuel assembly is blocked and the fuel assembly melts while in the pool. A fraction of the fuel assembly fission product inventory is released into the pool water and a fraction of this inventory is released into the air of the reactor hall. In a further step, a part of the total fission product content of reactor hall air is released to the environment by leakage from the reactor building. Effective (whole body) and thyroid doses are evaluated for this scenario for reactor building leak rates of 10% per hour and 20% per hour and compared with dose limits prescribed by Bulgarian law [11] and European Council directive [12].

The fission product inventory was calculated using the ORIGEN2.2 code [13] using burn-up and power distribution data from the neutronic analyses. A fuel assembly with an average ^{235}U burnup of 40% is used to determine the fission product inventory. It is assumed conservatively that the reactor worked continuously at 1 MW total power for 960 days.

From the calculated fission product inventory, the most important isotopes [14] which contribute to the doses were selected. The source term for radioactivity in the air of the reactor hall is determined as the inventory of one fuel assembly times: the transfer factor from fuel material → matrix material, the transfer factor from matrix material → water, and the transfer factor from water → air. However, since specific factors for each of these transfers are not available, a combined factor [15] for transfer of fission products from the fuel matrix material to the air of the reactor building is used here.

A spreadsheet based on the methodology described in [16, 17] was used to calculate data for evaluation of the doses. The following assumptions were made for the input: 1) Release of fission products occurs in a single phase of 1 hour duration; 2) The release height for the fission products is ground level (0 m). The dimensions of the reactor building are: (height: 16 m, width: 35 m, length: 35 m; Volume = 19,600 m^3); 3) A conservative meteorological model [18] was used fixing the meteorological conditions to Pasquill stability class F with 1 m/s

wind speed with uniform direction for a time period of 0-8 hours. Additionally, a Pasquill stability class F with a wind speed of 1 m/s with a variable direction within a 22.5˚ sector for a time period of 8-24 hours was utilized; 4) For distances below 100 m from the IRT-Sofia reactor, the atmospheric dispersion in air and dose values are identical to the corresponding values at 100 m; 5) It is assumed that the ventilation system is shut down at the time of the accident [7], so that no credit is taken for the reactor filtration system.

The results included in this evaluation are: a) doses for exposure of the last staff member leaving the reactor hall after the postulated accident (it is assumed that the last staff member evacuates the reactor hall after 5 minutes); b) doses for exposure of a member of the public present at the INRNE site fence (the location closest to the IRT, Sofia reactor is at a distance of about 100 m, and it is assumed that the person stays there for 2 hours); c) doses for exposure of the member of the public living closest to the IRT, Sofia (the closest permanently inhabited house is approximately 300 m from the IRT, Sofia and the wind is assumed to blow in the direction of the closest house; the doses are computed for 24 hours, which is sufficient to consider all effects due to inhalation). These results are summarized in Table 1.

Table 2.1 Comparison of Calculated Doses[a] with Dose Limits

Exposed Individual	Exposure Time	Effective (Whole Body) Dose			Thyroid Dose		
		Calculated Dose, mSv	Dose Limits, mSv		Calculated Dose, mSv	Dose Limits, mSv	
			Bulgarian Law	European Council Directive		Bulgarian Law	European Council Directive
Maximum Exposed Worker	5 min.	4.4	50 / year	50 / year	8.9	1000 / year	1000 / year
Maximum Exposed Member of Public	2 hours	0.3[b] 0.5[c]	1 / year	1 / year	0.8[b] 1.5[c]	20 / year	20 / year
Maximum Exposed Permanent Resident	24 hours	0.6[b] 0.9[c]	1 / year	1 / year	2.7[b] 3.6[c]	20 / year	20 / year

[a] Doses and Dose Limits are committed dose equivalents (CDE), which are calculated for individual organs and tissues over a 50-year period after inhalation [19].

[b] For building leak rate of 10% per hour.   [c] For building leak rate of 20% per hour.

Comparison of the calculated results with the dose limits shows that the calculated doses are below the dose limits which are specified by Bulgarian law and European Council directive for building leak rates of 10% per hour and 20% per hour.

One recommendation is to determine site specific weather conditions to properly assess the doses to the public receptors – the maximum exposed member of the public and the maximum exposed permanent resident. In general, when site specific weather conditions are used, the doses are less than the conservative values used for atmospheric conditions assumed.

7. Conclusions

The results of the presented accident analyses will be included in to the next version of the IRT, Sofia Safety Analyses Report. These results demonstrate the IRT, Sofia Safety System capability to work properly providing safe operation during the accidents analyzed.

The results obtained for Beyond Design Basis Accident shows that the calculated effective (whole body) and thyroid doses for exposed workers and the members of the public are below the limits set by the Bulgarian Law and the European Council directive at any location inside

or outside of the reactor building even in the frames of the considered conservative approach. These results reevaluation for more realistic meteorological conditions will be done soon.

We intend to continue the fruitful joint study between INRNE and the RERTR Program at ANL for successful completion of the safety analysis for the selected LEU core and Safety Analyses Report preparation for IRT, Sofia in accordance with licensing requirements.

8. Acknowledgement

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9. References

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