COMPARATIVE DOSE CALCULATION FOR TRIGA HEU AND LEU FUEL IN NUCLEAR ACCIDENT SITUATIONS

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Abstract. A 14 Mw TRIGA research reactor is operated on the Institute for Nuclear Research site. In the event of a nuclear accident or radiological emergency that may affect the public the effectiveness of protective actions depends on the adequacy of the selected accident scenarios and estimation of radiological consequences to public and environment, prepared in advance. The paper present comparative calculation for two accident scenarios and for both for HEU and LEU nuclear fuel. The evaluation of the radiological consequences considers both early and late consequences.

INTRODUCTION

The Institute for Nuclear Research (INR) Pitesti is located at 20 km far from Pitesti city and 5 km from Mioveni (NE). It is the largest Institute in Romania, whose main role is to develop research products and services to ensure technical support for nuclear power in Romania.

The Research Reactor facility in Romania is a dual core TRIGA reactor containing a 14MW TRIGA for steady-state operation and an ACPR TRIGA for pulse operation until 20,000 MW pulses. Both reactor cores are installed in a large pool containing 300 m3 of demineralized water, connected to the primary cooling system (see Figure 1).

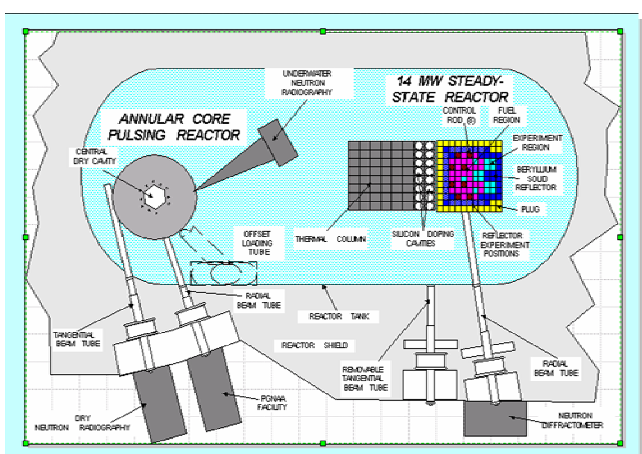

FIG. 1. The layout of the dual core TRIGA Research Reactor

The 14MW TRIGA R.R. is a unique design of TRIGA conception. Both reactor cores have individual Safety Analysis Reports, operational procedures, licensed operators and are authorized by the regulatory body, CNCAN, for continuous operation.

The Steady State core was fully converted in May 2006 to use LEU fuel. The core contains 29 fuel assemblies, 8 control rods and beryllium reflector, associated instrumentation and controls. The Annular Core Pulsed Reactor (ACPR) TRIGA is fueled for life. This reactor is mainly used at a low power level, i.e., 500 kW max. for NAA, beam application and primarily in the pulse mode to simulate transients and accident conditions RIA and LOCA type, when fuel consumption is not significant. This reactor is also used for training, education and demonstrations.

The decision to initiate a protective action in the case of nuclear accident or radiological emergency is a complex process. The benefits of taking the action is weighed against the risk and constraints involved in taking the action. In addition the decision will be made under difficult emergency conditions, probably with little detailed information available. Therefore, considerable planning is necessary to reduce to manageable levels the types of decisions leading to effective responses to protect the public in the event of a nuclear incident.

The sequence of events for developing emergency plans and responding to nuclear incidents [1, 2] will vary according to individual circumstances, because the international recommendations and site-specific emergency plans cannot provide detailed guidance for all accident scenarios and variations in local conditions. Flexibility must be maintained in emergency response to reflect the actual circumstances encountered [3] (e.g. source term characteristics, the large number of possible weather conditions and environmental situations such as time of the day, season of the year, land use and soil types, population distribution and economic structures, uncertainties in the availability of technical and administrative support and the behavior of the population). This further complicates the decision-making process, especially under accident conditions where there are time pressures and psychological stress.

Therefore one of the most important problems in the case of a nuclear emergency is quantifying all these very different types of off-site consequences.

ACCIDENT SCENARIOS

For the Steady State core of TRIGA Research Reactor were identified [4,5] both internal and external events leading to an emergency situation. The accident scenarios considered for design basis accidents and beyond design basis accidents are presented in Table 1.

Table 1. The accident scenarios considered for the steady state core

Accident	Description	Release Conditions
Anticipated release	Single pin failure in water with experimentally determined release fractions from TRIGA fuel	Fraction of core involved: 1 pin (0.14% of core) Fraction of fission products available for the release: 6.3×10^{-4} Fraction of available fission products released to the pool: - noble gases 100% - halogens 25% Fraction of fission products released from the pool water: - noble gases 100% - organic halogens 25% - elemental and particulate halogens (90% of total) 1% Condition of ventilation system: normal Exhaust rate from stack: 24,360 m^3/h

Table 1. The accident scenarios considered for the steady state core (continued)

Accident	Description	Release Conditions
Design basis release	25-pin failure in air with total release of volatile fission products from TRIGA fuel	Fraction of core involved: single bundle (3.4% of core) Fraction of fission products released from fuel to reactor hall: - noble gases 100% - organic halogens 25% Condition of ventilation system: emergency Exhaust rate from stack: 6000 m^3/h
Modified design basis release (1)	25-pins failure in water with experimentally determined release fractions from TRIGA fuel	Fraction of core involved: 3.4% of core Fraction of fission products available for release from fuel: 6.3×10^{-4} Fraction of fission products released from pool water: - noble gases 100% - organic halogens 100% - elemental and particulate halogens (90% of total) 1% Condition of ventilation system: emergency Exhaust rate from stack: 6000 m^3/h
Modified design basis release (2)	25-pins failure in water with total release of volatile fission products from TRIGA fuel	Fraction of core involved: 3.4% of core Fraction of fission products released from fuel: - noble gases 100% - halogens 25% Fraction of fission products released from pool water: - noble gases 100% - organic halogens 100% - elemental and particulate halogens (90% of total) 1% Condition of ventilation system: emergency Exhaust rate from stack: 6000 m^3/h

For detailed calculations two accident scenarios were selected: 25-pin failure in air with total release of volatile fission products from TRIGA fuel – **Scenario No.1**, and 25-pins failure in water with experimentally determined release fractions from TRIGA fuel – **Scenario No.2**.

The failure of a 25-pin fuel bundle (due to mechanical damage) with the consequent release of fission products is an event that has a small but significant probability and, over the life of the core.

Two modified design basis release conditions have been, also, analyzed. The modified design basis release considers the fuel pins failure in water and uses an experimentally determined [6] fission products release fraction for the fuel-moderator material.

These failures were analyzed using the following assumptions:

a) The fuel pins that fail have operated at an average power density of twice as great as the average power density in the core;

b) The core has operated continuously for a total of 7700 MWd;

c) During the accident evolution, the emergency ventilation system and charcoal traps are not available, so no fission products will be retained by traps

d) The release height is assumed to be 50 m above the ground level;

e) The radiological consequences assessment has been performed with PC-COSYMA computer code [7,8], considering a site specific meteorological file and site specific databases.

The noble gases and halogen fission products inventory was calculated [5, 6] assuming a burnup of 7700 MWd occurring in 1.5 calendar years. The bundle that was assumed to have failed operated at a power density twice that of the average in the 29-bundle core (i.e., at 970 kW/bundle). It is assumed that a fraction of the ***i**th* isotope from this inventory was released to the reactor hall instantaneously. This fraction $\boldsymbol{w_i}$ is:

$$w_i = \left(\frac{p}{N}\right) \cdot e_i \cdot f_i \cdot g_i$$

where: (p/N) = the relative power density in the failed bundle **= 2/29**

e_i = the fraction released to the fuel-clad gap

f_i = the fraction of the $\boldsymbol{i^{th}}$ isotope released to the pool

g_i = the fraction of the $\boldsymbol{i^{th}}$ isotope released to the reactor room

For the anticipated release the value of e_i is 6.3E-04, whereas for the design basis release, it is assumed to be equal to 1. For release in water, the values for f_i and g_i are shown in Table 2.

Table 2. Values for the f_i and g_i parameters

Fission product	$\mathbf{f_i}$	$\mathbf{g_i}$
Noble gases	1.00	1.00
Halogens	0.25	0.109 =0.1+(0.1x0.9)
Others	0.00	0.00

The value of g_i for the halogens arises from the assumption that 10% of the halogens form organic compounds, insoluble in water, and 90% of the halogens are in elemental or particulate form of which all but 1% are retained in the water. For the analysis of release in air the value of g_i for the halogens is also 1.

For the releases of the fission products from the building, two modes were considered: removal by the emergency ventilation system through an activated charcoal trap, and removal by the normal ventilation system, respectively. The first of these modes is the one for which the system is designed. In the analysis, the charcoal filters are assumed to have an efficiency of 0% for noble gases and 99% for the halogens.

In the last years, and in particular since the Chernobyl accident, there has been a considerable increase in the resources allocated to development of computerized systems which allow for predicting the radiological impact of accidents and to provide information in a manageable and effective form to evaluate alternative countermeasure strategies in the various stage of an accident. In Table 3, are summarized the components of the computerized support for nuclear accident management.

Table 3. The computerized support for nuclear accident management

Computer codes	Site Specific Databases
DOZIM (developed in ICN)	Population
COSYMA (EU)	Meteorological
MACCS (USA)	Agricultural production
	Animal production
	Food consumption rates

RESULTS

The evaluation of consequences to population and environment were performed considering that no protective measure was implemented.

The isotopic inventory was obtained with ORIGEN-S computer code [9,10] with above assumption for power density and burnup.

The comparative results for uranium, neptunium, plutonium and americium isotopic inventories are presented in Figure 2.

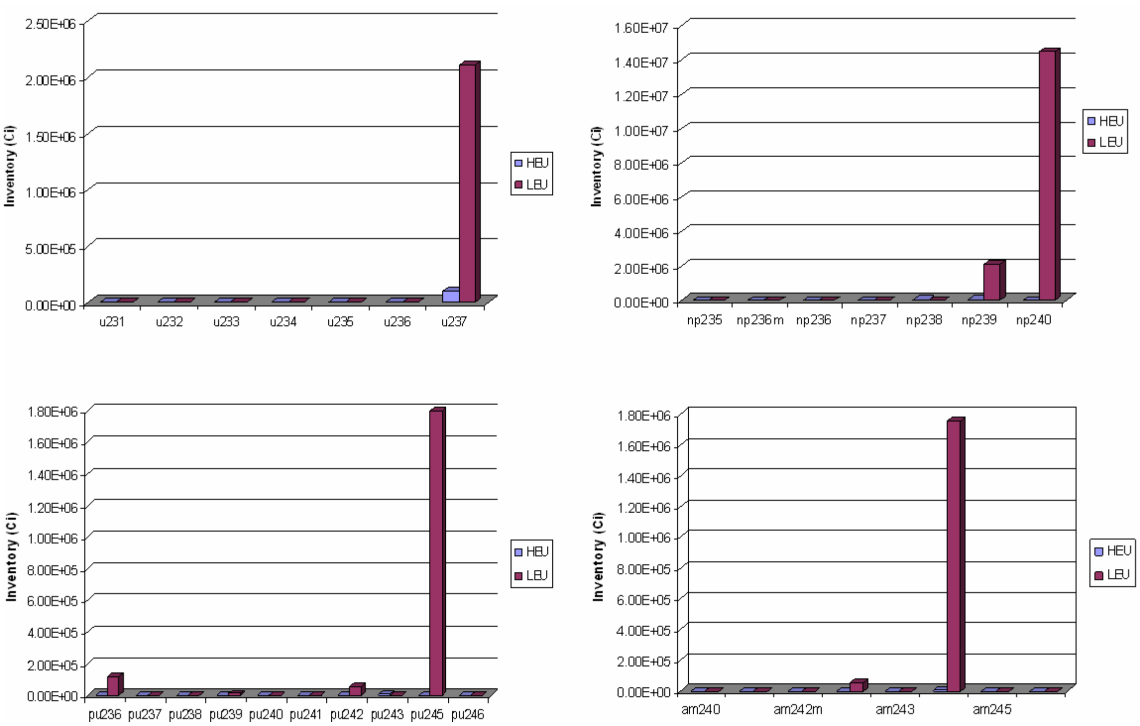

FIG. 2. The comparative results for uranium, neptunium, plutonium and americium isotopic inventories

The variation of ground and air concentrations for isotope I-131 for both accident scenarios is presented in Figure 3 and Figure 4, respectively.

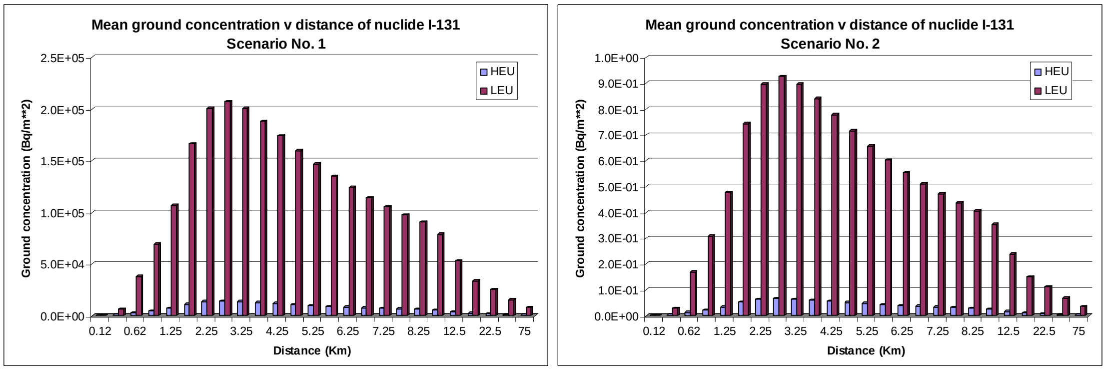

FIG. 3. The variation of ground concentrations for isotope I-131 for both accident scenarios

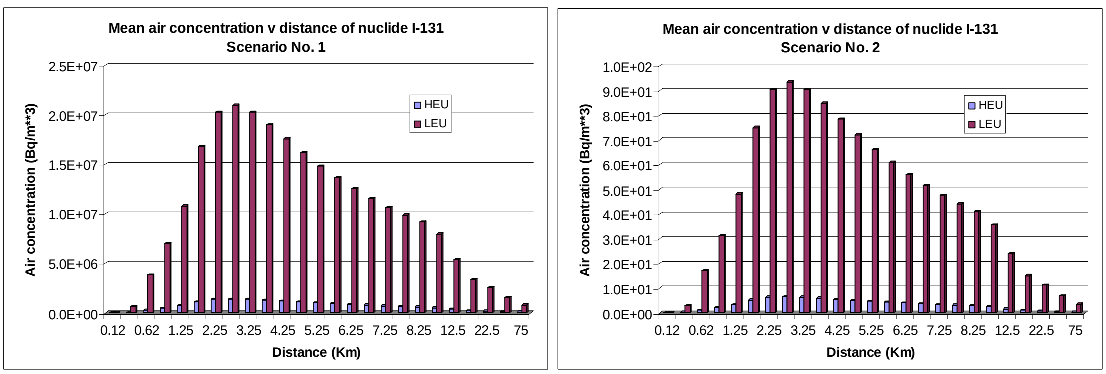

FIG. 4. The variation of air concentrations for isotope I-131 for both accident scenarios

The residence most close to Institute for Nuclear Research site is located around 2 kilometers in straight line, this limit representing the approximate boundary to the residential area of city of Mioveni (approximate 30,000 inhabitants), so in Figure 5 and 6 are presented the ground and air concentrations at 2.250 Km far from site. The variation of mean individual 1 day effective dose and mean long term individual dose at 50 years with distance for both accident scenarios is presented in Figure 7 and Figure 8 respectively.

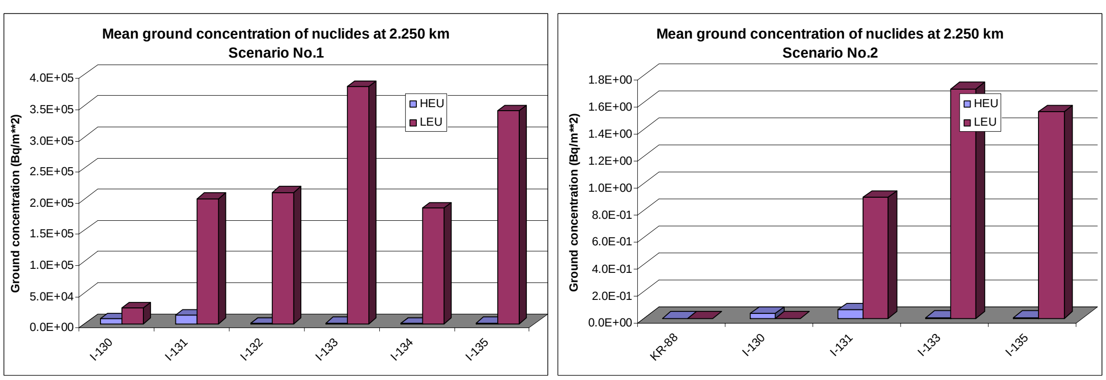

FIG. 5. The variation of ground concentrations at 2.250 Km far from site

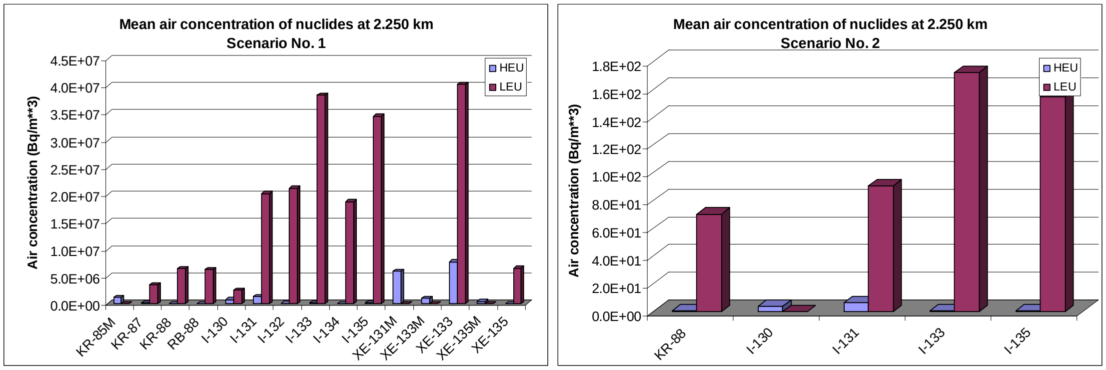

FIG. 6. The variation of air concentrations at 2.250 Km far from site

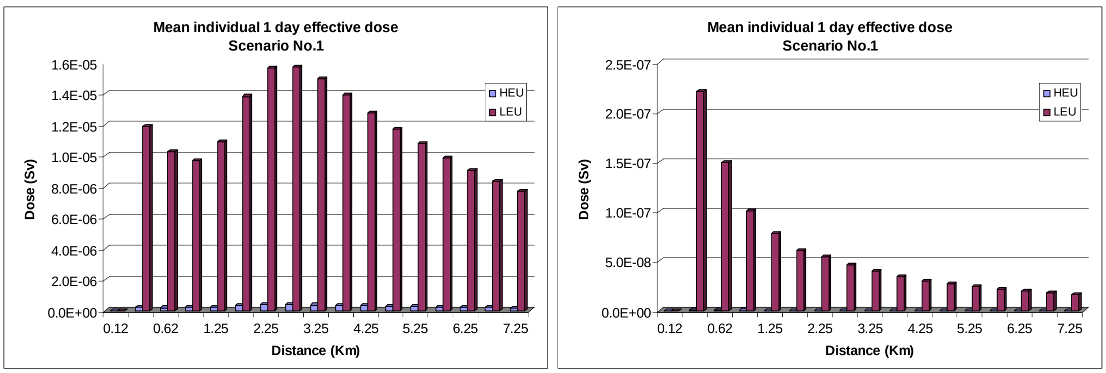

FIG. 7. The variation of mean individual 1 day effective dose with distance for both accident scenarios

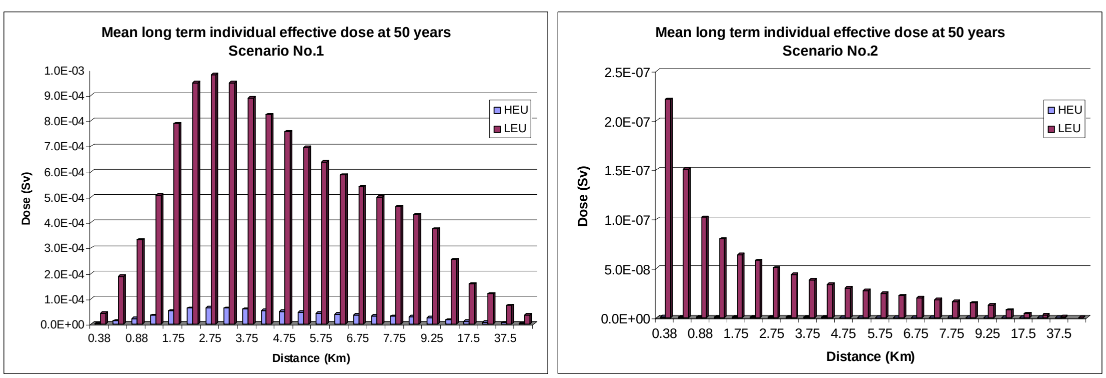

FIG. 8. The variation of mean long term individual dose at 50 years with distance for both accident scenarios

The collective doses to 50 years emphasizing the pathways involved are presented in Figure 9. As a result of the radioactive contamination of the environment, a certain number of early and late health effects in population could occur. The total number of health effects for both accident scenarios are shown in Figure 10.

Organ	Collective Dose (manSv)		Pathway									
			Cloud		*Ground*		*Inhalation*		*Ingestion*		*Resuspension*	
	HEU	LEU	HEU	LEU	HEU	LEU	HEU	LEU	HEU	LEU	HEU	LEU
Scenario No.1												
B.MARROW	4.41E-01	8.60E+00	2	6	80	79	2	2	16	13	0	0
B.SURFACE	4.83E-01	9.33E+00	3	7	77	76	2	2	19	14	0	0
BREAST	2.08E-01	4.13E+00	3	8	86	83	1	2	10	8	0	0
LUNG	4.69E-01	9.30E+00	2	6	79	77	3	5	16	12	0	0
STOMACH	5.70E-01	1.10E+01	2	5	58	58	2	6	38	32	0	0
COLON	4.51E-01	8.62E+00	2	6	71	71	1	2	26	21	0	0
LIVER	3.80E-01	7.56E+00	3	7	87	84	1	2	9	7	0	0
PANCREAS	3.62E-01	7.19E+00	3	7	84	81	1	2	12	9	0	0
THYROID	3.27E+02	4.99E+03	0	0	0	0	6	7	94	92	0	0
GONADS	3.88E-01	7.72E+00	3	7	88	85	1	2	8	7	0	0
REMAINDER	5.65E-01	1.10E+01	2	6	65	64	3	6	30	24	0	0
EFFECTIVE	1.68E+01	2.59E+02	0	0	2	3	6	7	92	90	0	0
Scenario No.2												
B.MARROW	1.44E-05	3.23E-03	79	91	13	2	6	7	2	0	0	0
B.SURFACE	1.50E-05	3.34E-03	78	90	13	2	6	7	3	0	0	0
BREAST	7.85E-06	1.81E-03	82	92	12	2	5	6	1	0	0	0
LUNG	2.90E-05	6.99E-03	40	43	7	1	52	55	1	0	0	0
STOMACH	4.05E-05	9.87E-03	28	29	4	1	66	70	3	0	0	0
COLON	1.39E-05	3.09E-03	77	90	12	2	7	7	4	0	0	0
LIVER	1.40E-05	3.21E-03	81	91	12	2	6	6	1	0	0	0
PANCREAS	1.34E-05	3.06E-03	79	90	12	2	7	8	2	0	0	0
THYROID	1.56E-03	2.62E-02	1	14	0	0	6	7	93	79	0	0
GONADS	1.36E-05	3.08E-03	80	91	13	2	6	6	1	0	0	0
REMAINDER	1.60E-05	3.51E-03	77	90	12	2	6	7	5	0	0	0
EFFECTIVE	9.93E-05	6.26E-03	12	49	2	1	12	33	74	17	0	0

FIG. 9. The collective doses to 50 years by pathway

Organ	Numbers of late health effects							
	Mortality		*Incidence*		*Mortality*		*Incidence*	
	HEU	LEU	HEU	LEU	HEU	LEU	HEU	LEU
	Scenario No.1				**Scenario No.2**			
bone mar.	2.28E-03	4.44E-02	2.28E-03	4.44E-02	7.41E-08	1.67E-05	7.41E-08	1.67E-05
bone surf.	6.42E-05	1.24E-03	6.42E-05	1.24E-03	1.99E-09	4.45E-07	1.99E-09	4.45E-07
breast	3.33E-03	6.61E-02	8.32E-03	1.65E-01	1.26E-07	2.89E-05	3.14E-07	7.22E-05
lung	4.22E-03	8.37E-02	5.63E-03	1.12E-01	2.61E-07	6.29E-05	3.47E-07	8.38E-05
stomach	5.16E-03	9.94E-02	6.07E-03	1.17E-01	3.67E-07	8.93E-05	4.31E-07	1.05E-04
colon	1.55E-03	2.96E-02	2.81E-03	5.38E-02	4.76E-08	1.06E-05	8.65E-08	1.93E-05
liver	1.77E-03	3.53E-02	1.77E-03	3.53E-02	6.55E-08	1.50E-05	6.55E-08	1.50E-05
pancreas	1.90E-03	3.78E-02	2.12E-03	4.20E-02	7.03E-08	1.61E-05	7.81E-08	1.79E-05
thyroid	5.79E-01	8.84E+00	5.79E+00	8.84E+01	2.77E-06	4.64E-05	2.77E-05	4.64E-04
remainder	2.18E-03	4.23E-02	3.63E-03	7.05E-02	6.17E-08	1.36E-05	1.03E-07	2.26E-05
skin	1.60E-03	2.97E-02	1.60E-01	2.97E+00	8.42E-09	3.54E-07	8.42E-07	3.54E-05
hered.eff.	3.88E-03	7.72E-02	3.88E-03	7.72E-02	1.36E-07	3.08E-05	1.36E-07	3.08E-05
Total	**6.03E-01**	**9.30E+00**	**5.98E+00**	**9.20E+01**	**3.85E-06**	**3.00E-04**	**3.00E-05**	**8.52E-04**

FIG. 10. The total number of health effects

Unfortunately, data about the behavior during accidents, fractions of fission products released from fuel to pool water and from pool water to air in reactor hall for the LEU nuclear fuel are not available; this is why in calculations we considered the same release factors both for HEU and LEU fuel. As can be seen, the values for radioactive concentrations in environment and doses to public are small, so even in these accident situations, the consequences for the public and environment will be very small.

CONCLUSIONS

The paper briefly presents the consequences of two nuclear accident scenarios for 14 MW TRIGA Research Reactor from Institute for Nuclear Research Pitesti. A nuclear accident is a dynamic phenomenon in space and time, and its evolution can be predicted with a certain probability.

Moreover, a real nuclear accident will require a realistic evaluation of the nuclear installations and safety systems status, leading to detailed information about the conditions before and during the accident and also the amount of the released radioactive contaminants.

Unfortunately, more likely, such kind of information is not available in a short time after the accident initiation. For this reason, detailed analyses, based on various nuclear accidents scenario, obtained from the PSA studies, are needed.

ACKNOWLEDGEMENTS

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