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Radiological Safety Analyses for MNSR LEU Conversion Study

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ABSTRACT

Radiological safety analyses are an integral part of the overall safety analysis for conversion of Miniature Neutron Source Reactors (MNSRs) to low-enriched uranium (LEU) fuel. Calculations have been performed for generic MNSR cores to derive peak radionuclide activities in both the high-enriched uranium (HEU) cores and the LEU cores using the ORIGEN2 code. Nuclide inventories were used to calculated inhalation doses as a result of a postulated design basis accident for workers and the public. Photon spectra from the ORIGEN2 calculations were also used to calculate direct gamma-ray dose rates following a postulated loss-of-pool-water accident and following a beyond design basis accident. Comparison of the results for the two core types shows that dose rates for postulated accidents with the LEU core are approximately 20% higher than those for the HEU core, due to the longer LEU core lifetime and higher operating power. Calculated doses for both cores remain within regulatory limits.

1. Introduction

This paper provides information on radiological safety-related aspects of the Miniature Neutron Source Reactor (MNSR) LEU core conversion analyses. The analyses are based on the established models of generic HEU and LEU cores for MNSRs developed by Argonne National Laboratory as part of the Global Threat Reduction Initiative (GTRI) - Conversion Program.

The generic HEU core model (denoted HEU345) with 345 fuel pins was established from existing HEU models for the NIRR-1 reactor [1] in Nigeria and the GHARR-1 reactor [2] in Ghana. The main purposes for using this generic HEU model are two-fold: (i) it can be easily adapted by each individual commercial reactor operator to model his own HEU MNSR reactor;

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and (ii) it can also be easily modified to establish a generic LEU core that will serve as the base model for LEU conversion studies of individual commercial MNSR reactors.

In this study, a base generic 345-pin HEU model was first defined and then augmented to form as-built, working cores for the GHARR-1 and NIRR-1 reactors. A generic LEU model (denoted LEU348) with 348 fuel pins was then defined using the specifications that were agreed to by the MNSR community at the 2nd RCM in Vienna in May 2008 [3]. An enrichment of 12.5% was chosen so that the HEU and LEU cores have approximately the same number of fuel pins, similar water-to-metal ratios, and hence, about the same negative reactivity feedback and power coefficients.

Radiological safety analyses for the MNSR LEU conversion study involves calculation of the radioactive nuclide inventory and determination of the maximum radiation source terms (gamma and neutron spectra) during the core lifetime. Subsequently, based on the maximum source terms, radiological doses are calculated at various locations surrounding the reactor core under various operational conditions and accident scenarios. Results are then analyzed and compared against regulatory limits for the licensing of the reactor to demonstrate radiation safety.

2. Radionuclide Activities in Peak Burnup Fuel Rods for MNSR Generic Cores

Calculations were performed to obtain the inventory of halogens, noble gases, alkaline metals, and actinides in the peak power fuel rods for the generic MNSR HEU345 and LEU348 cores operated at typical power levels. The fuel meat of the HEU345 fuel rods are U-Al alloy and the fuel meat of the LEU348 fuel rods are UO_2. The inventory data reported here can be used for radiological assessment of accident scenarios involving the release of radioactive material (e.g., maximum hypothetical accidents, or MHAs).

Pertinent design data for the HEU345 and LEU348 fuel rods are summarized in Table 1. The LEU348 core is designed to have a higher total core power (in order to maintain the same thermal flux levels at irradiation channels), and has a slightly larger peak-to-average rod power than the HEU345 core. The peak rod powers used for the inventory analysis are 99.66 and 113.35 W for the generic HEU345 and LEU348 cores, respectively.

Inventory data were calculated with ORIGEN2 [4] code package, a zero-dimensional isotope decay and transmutation code. The code solves the transmutation equations using libraries of radioactive decay data and 1-group cross section data. The base code libraries allow the tracking of over 100 actinides and nearly 900 fission product nuclides. Pre-calculated libraries of 1-group cross section data are available for use in ORIGEN2 for several reactor systems. A standard library for an oxide-fueled LWR, the so-called PWRUS library, represents the closest potential match to data for the generic MNSR cores. The anticipated core residence time during irradiation period is measured in Full-Power-Equivalent-Days (FPEDs) or Half-Power-Equivalent-Days (HPEDs).

Table 1. Generic MNSR HEU and LEU Fuel Rod Parameters

Parameter	HEU	LEU
Reactor Full Power, kW	30.0	34.0
Number of Rods in Core	345	348
Uranium Density in Fuel Meat, g/cm^3	0.92	9.35
Enrichment, %	90.2	12.5
Fuel Meat O.D., mm	4.3	4.3
Fuel Rod O.D., mm	5.5	5.5
^{235}U/Rod	2.8721	3.8915
^{238}U/Rod	0.2872	27.1007
P_{max}/P_{avg} Rod Power	1.15	1.16
Peak Rod Power, W	99.66	113.35
Anticipated Core Residence, FPE Days	810	903
Anticipated Core Residence, HPE Days	1,614	1,800

For the present analysis, it is preferable to replace certain cross section data with data that are more appropriate to the MNSR system being analyzed. Replacement 1-group cross section data were calculated for about a dozen selected actinides and fission products for the MNSR fuel rods using the WIMS-ANL [5] code. The calculated 1-group capture and fission cross sections for the HEU345 and LEU348 rods are compared with the cross section data from the ORIGEN2 PWRUS library in Table 2.

The ORIGEN2 code calculates radioactive nuclide inventories for three material groups, i.e., activation products, actinides and daughters, and fission products. The radioactive nuclide activities (in curies) are produced both during the irradiation period and cooling period afterward. For a bounding hypothetical accident radiological dose evaluation, the maximum value of the radionuclide activities over the whole core life history (including both irradiation period and cooling period) must be used.

Table 3 summarizes the bounding maxima of halogen, noble gas, alkaline metal, and actinide activities in the HEU345 and the LEU348 fuel rods continuously operating at their peak rod power. The column on the right-hand side of Table 3 shows ratios of the HEU345 results relative to those for LEU348. The higher fission product activities for halogens, noble gases, and alkaline metals in the LEU348 fuel rods relative to the HEU345 fuel rods are primarily due to the 13% increase in power level in the LEU348 design. These bounding radioactivities were determined from the detailed ORIGEN2 output data over irradiation and cooling periods for the peak power pin. The bounding maximum radioactivity values are almost always found near the beginning (75 FPEDs for HEU345 core and 90 FPEDs for LEU348 core) of the irradiation period for halogens (except for Br-82, I-130, and I-130m) and noble gas (except for Kr-85) fission products. However, the bounding maximum activities for alkaline metals and actinides and daughters are almost always found at the end of irradiation period (810 FPEDs for HEU345 core and 910 FPEDs for LEU348 core).

Table 2. One-Group Cross Section Data for Generic MNSR Fuel Rods

Nuclide	Capture			Fission		
	PWRUS Library[1]	Replacement Data[2]		PWRUS Library	Replacement Data[2]	
		HEU345	LEU348		HEU345	LEU348
U234	20.710	29.775	22.663	0.452	0.566	0.576
U235	10.680	15.891	12.247	47.520	80.711	59.750
U236	8.348	9.065	7.433	0.191	0.336	0.328
U238	0.887	5.189	1.138	0.093	0.114	0.119
NP237	33.280	46.924	39.706	0.495	0.530	0.559
PU238	34.830	68.997	49.836	2.308	3.373	2.867
PU239	69.090	74.765	60.520	121.100	151.830	119.050
PU240	222.800	260.670	241.630	0.579	0.619	0.647
PU241	42.020	60.101	45.743	125.900	174.790	133.020
PU242	33.200	31.004	29.887	0.406	0.448	0.472
AM241	95.700	149.000	123.000	1.120	1.367	1.254
XE135	221,500	420,000	350,000	0	0	0

[1]PWRUS library is a standard PWR library available with the ORIGEN2 code package.
[2]Replacement cross sections for HEU rod calculated with WIMS-ANL at 400 FPEDs operation; for LEU rod at 450 FPEDs operation

Table 3. Maximum Activities (GBq) for Generic MNSR Peak Power Fuel Rods Over the Entire Core Life Time

Radioactive Nuclides	HEU345	LEU348	LEU / HEU
Total Halogens	1.27×10^3	1.44×10^3	1.14
Total Noble Gases	1.20×10^3	1.37×10^3	1.14
Total Alkalines	2.21×10^3	2.52×10^3	1.14
Total Actinides	9.51×10^0	4.88×10^2	51.36

The actinide activity in the LEU348 rod is a factor of 51 higher than that in the HEU345 rod. As shown in Table 1, the U-238 loading in the LEU348 rods is a factor of 87 higher than that in the HEU345 rod. U-238 undergoes a neutron capture to form U-239, which quickly decays by β^- emission to Np-239. Subsequent transmutations of Np-239 lead to even higher actinides. The increase in the total higher actinide activity at the end of core life (discharge) is mostly due to the larger initial U-238 concentration in the LEU348 rods. The major contributors (Np-239 and U-237) to the actinide activities at end of irradiation are short-lived radionuclides with half-lives on the order of days. Within a few weeks after shutdown, these nuclides will have largely decayed to longer-lived Pu-238 and Pu-239. For the purpose of assessing the bounding radiological consequences of a release of actinides from fuel material, the maximum doses would be obtained at the time of discharge. Unless the hypothetical accident scenario assumes that it happens right at the precise moment at discharge, it should be reasonable to consider some level of cooling has occurred for accidents involving any kind of operational core access events or maintenance activities. Therefore any level of cooling assumed after discharge would be helpful to reduce the radiological dose levels due to the decay of these short lived actinides. The total actinide inventory in the LEU348 peak power rod decreases by nearly a factor of 500 from the discharge value after one month of post-irradiation cooling. It is also noted that the differences in the actinide inventory between the HEU345 and LEU348 peak power rods decreases with cooling. Consequently, any differences in hypothetical radiological dose evaluations between HEU345 core and LEU348 core would be reduced at longer cooling times.

3. Photon Spectra for Hypothetical Accident Dose Evaluation

The ORIGEN2 code produced 18-group photon spectra for three material groups, i.e., (i) activation products, (ii) actinides and daughters, and (iii) fission products both during the irradiation period and cooling period afterwards. For a bounding hypothetical accident dose evaluation, the maximum values of the photon spectra over the whole core life history (including both irradiation period and cooling period) must be used. The bounding maximum photon spectral values are almost always found at the end of the irradiation period for activation products and fission products. Some exceptions were found for the actinides and daughters where the maximum photon values in three fairly high energy groups (between 2.25 to 3.5 MeV) are found after a 10 - 20 year cooling period.

Typical photon source buildup during the irradiation period and then decay during the cooling period can be illustrated in Figure 1 for the HEU345 peak power pin. The shapes of the curves for the LEU348 peak power pin are similar. In both cases, the total photon source reaches its maximum at the end of irradiation period (discharge time) and then decays quickly during the initial cooling period of 30 days. After that the decay rate slows down progressively. Since almost 99% of the photon energy is produced by the fission products, the conservative bounding radiological dose due to hypothetical accidents can be evaluated at the end of irradiation (discharge time) when the photon source is at its maximum. Any level of cooling assumed after discharge would reduce the radiological dose levels due to the decay of these short lived fission products.

The corresponding bounding 18-group total photon spectra at the end of irradiation (discharge time) are shown in Table 4. These two photon spectra at the end of irradiation (discharge time) are used to obtain the bounding (maximum) radiological dose rates. It should be noted that these 18-group photon data are only applicable to external exposures (submersion doses). The inventory results of Section 2 for the maximum concentration of each isotope must be used to calculate internal exposures (inhalation doses) due to radioactive nuclides released from core and that the sum of the submersion plus inhalation doses gives the worst-case hypothetical accident dose result.

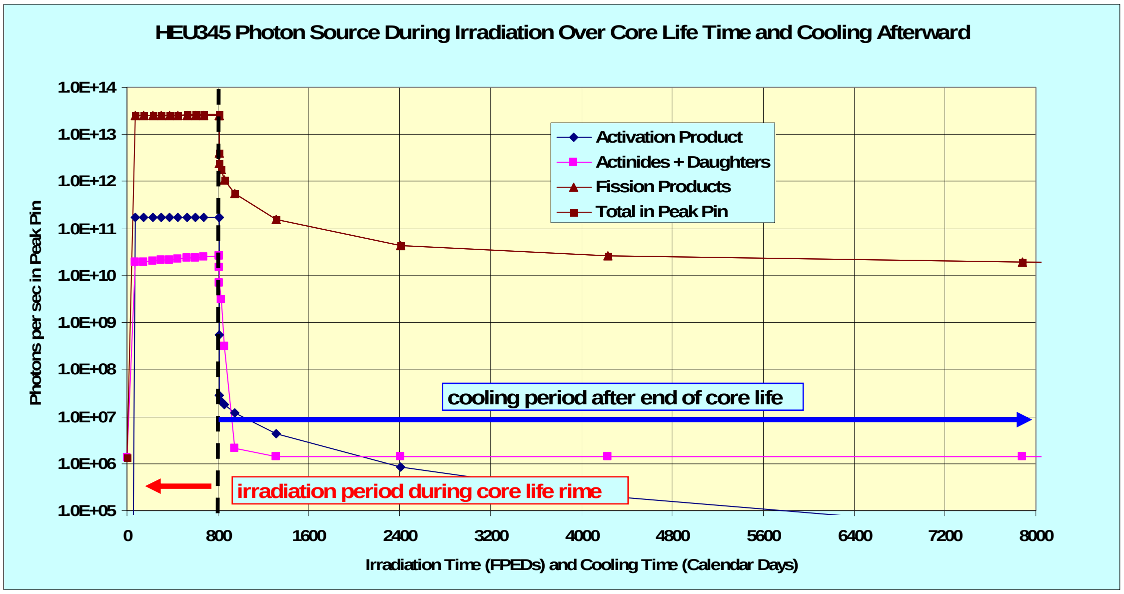

Figure 1. HEU345 Peak Power Pin Photon Source During Irradiation Period for 810 FPEDs and Cooling Period Afterward for 8,000 Calendar Days

Table 4. Gamma (Photon) Sources for Peak Power Pin at End-of-Core-Life in the HEU345 and the LEU348 Cores

Photon Group	Mean, MeV	HEU345 Source Photons/s	LEU348 Source Photons/s	LEU/HEU Ratio	Photon Group	Mean, MeV	HEU345 Source Photons/s	LEU348 Source Photons/s	LEU/HEU Ratio
1	1.00E-02	6.61E+12	8.13E+12	1.23	10	8.50E-01	2.95E+12	3.37E+12	1.14
2	2.50E-02	1.65E+12	1.90E+12	1.15	11	1.25E+00	1.77E+12	2.01E+12	1.14
3	3.75E-02	1.38E+12	1.61E+12	1.17	12	1.75E+00	7.02E+11	7.22E+11	1.03
4	5.75E-02	1.42E+12	1.64E+12	1.16	13	2.25E+00	3.54E+11	4.02E+11	1.14
5	8.50E-02	1.05E+12	1.49E+12	1.42	14	2.75E+00	1.52E+11	1.72E+11	1.13
6	1.25E-01	1.09E+12	1.48E+12	1.36	15	3.50E+00	9.17E+10	1.04E+11	1.13
7	2.25E-01	2.55E+12	3.06E+12	1.20	16	5.00E+00	4.95E+10	5.57E+10	1.13
8	3.75E-01	1.53E+12	1.77E+12	1.16	17	7.00E+00	3.99E+08	4.56E+08	1.15
9	5.75E-01	2.61E+12	2.98E+12	1.14	18	9.50E+00	7.88E+04	9.40E+04	1.19
						TOTAL	**2.60E+13**	**3.09E+13**	**1.19**

4. Loss of Pool Water Shielding Accident

In this accident scenario, described in Refs. [1] and [2], a major earthquake is assumed to cause a crack on the bottom floor of the pool, resulting in the pool water draining below the level of the core. Simultaneous loss of water in the reactor vessel is not considered to be credible because the vessel is designed and constructed to support the core while suspended in an empty pool. The reactor is assumed to continue operating, but the power level will decrease because of the increasingly negative reactivity caused by loss of cooling water to the pool (i.e., negative temperature coefficient).

The loss of pool water will cause very high gamma radiation fields over the reactor. A simulation experiment [6] was carried out for the prototype MNSR at the CIAE near Beijing for the loss of pool water accident. Portable gamma monitors were placed at positions 1-4 (Figure 2) close to the lower end of the defense fence with the sensors sticking into the pool. A fixed gamma monitor was installed at position 5 and the sensor placed on top of the flange of the reactor and towards the bottom of the pool.

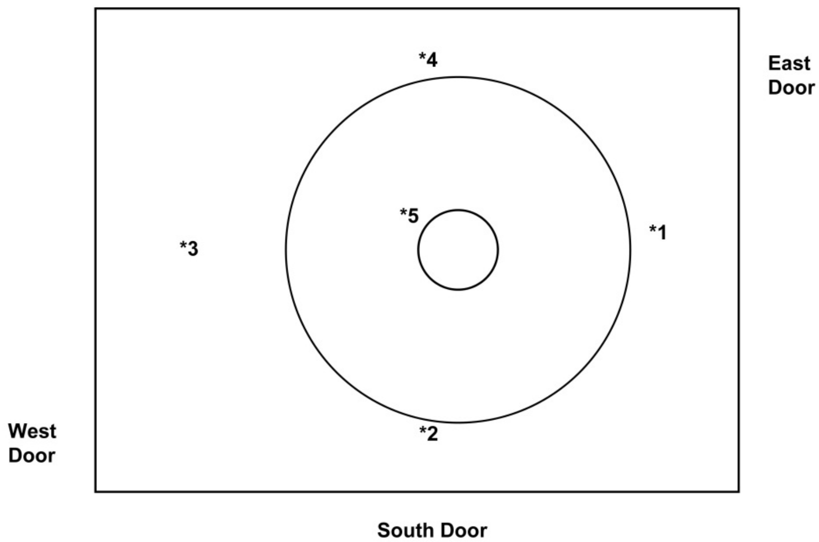

Figure 2. Positions for Gamma Monitoring for Simulating Loss-of-Pool Water Accident (from Ref. 6)

The measured gamma dose rates (mSv/h) for the different monitoring positions are shown in Table 5.

Table 5. Gamma Dose Rates (mSv/h) Measured at Monitoring Positions in the HEU MNSR Prototype Reactor and Expected Values for LEU Reactor

Position	1	2	3	4
Measured Value for HEU Core	4.0×10^{2}	4.2×10^{2}	4.2×10^{2}	6.2×10^{2}
Expected Value for LEU Core	4.8×10^{2}	5.0×10^{2}	5.0×10^{2}	7.4×10^{2}
Position	5	East Door	South Door	West Door
Measured Value for HEU Core	1.1×10^{2}	1.6×10^{-2}	2.3×10^{-1}	2.3×10^{-1}
Expected Value for LEU Core	1.3×10^{2}	1.9×10^{-2}	2.7×10^{-1}	2.7×10^{-1}

As shown in Table 4, the 18-group gamma (photon) spectra indicate that the gamma source for the LEU core is 1.19 times larger than that for the HEU core, because the LEU core has a higher power level and a longer estimated core lifetime. As a result, gamma dose rates in an LEU MNSR are expected to be about 1.19 times the dose rates that were measured in the HEU MNSR prototype reactor. These estimated dose rates in an LEU MNSR are shown in Table 5 for comparison. Some reduction in dose rates in the LEU core may be expected because gamma rays emanating from the LEU core need to pass through denser material (namely, UO_2 instead of UAl alloy).

The average dose rate at gamma monitoring positions 1-5 in Table 5 for the HEU MNSR Prototype reactor was measured as 4.0×10^2 mSv/h and is estimated to be about 4.7×10^2 mSv/h in the generic LEU core. During operation, the area within the fence on top of the reactor is the controlled area where no individual would be staying when the accident occurs.

The average dose rate at the east, south and west entrance doors of the reactor hall was measured to be 1.6×10^{-1} mSv/h in the HEU Prototype reactor (Table 5) and estimated to be 1.9×10^{-1} mSv/h in the generic LEU reactor. The exposure of individuals working for 8 hours on the main floor of the reactor hall or in the adjoining room during an emergency would be approximately 1.3 mSv in the HEU case and 1.5 mSv in the LEU case. Both of these doses are far below the recommended maximum effective (whole body) dose of 50 mSv / year in IAEA Safety Series 115 [7]. Since the reactor would be shut down, the actual dose would be much smaller. Thus it would not cause any risk to operating personnel.

5. Design Basis Accident / Maximum Credible Accident

As the Design Basis Accident (DBA) for the generic MNSR, it is postulated that pitting corrosion of the cladding creates cladding failure in one or more fuel rods such that a hole or holes in the cladding are formed totaling 0.5 cm^2 while in the water of the reactor vessel. A fraction of the fuel rod fission product inventory is released into the pool water and a fraction of this inventory is released into the air of the reactor hall. Furthermore, part of the total fission product content of air in the reactor hall is released to the environment by leakage from the reactor building. Effective (whole body) and thyroid doses are evaluated for this scenario for a reactor building leak rate of 20 volume% per hour and compared with dose limits recommended by IAEA Safety Series No. 115.

Source Term Determination

The source term for radioactivity in the air of the reactor hall is determined by the inventory of the fuel rod with peak ^{235}U burnup as described in Section 2, and the three transfer factors from fuel material to matrix material, from matrix material to water, and from water to air, respectively. However, since specific factors for each of these transfers are not available, a combined factor [8] for the transfer of fission products from the fuel matrix material to the air of the reactor building is used here. The fission product inventories in the fuel rod with peak ^{235}U burnup are taken from the results of the calculations described in Section 2. The combined transfer factors are: 1×10^{-4} for halogens, 0.02 for noble gases, 1×10^{-6} for alkalines, lanthanides, alkaline metals and actinides.

Evaluation of Radiological Consequences

A spreadsheet based on the methodology described in Ref. [9] was used to calculate data for evaluation of the doses. A close approximation to the doses calculated using this spreadsheet can be obtained using the basic equations and methodology described in Ref. [10]. The following assumptions were made for the input:

- Source term as determined in Section 2.
- Release of fission products occurs in a single interval of 1 hour duration.
- The release height for the fission products is ground level (0 m). The dimensions of the reactor building are: height = 8.5 m, width = 7.1 m, length = 7.2 m, volume = 434.52 m^3.
- A conservative meteorological model [11] 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 used.
- For distances below 100 m from the reactor, the atmospheric dispersion in air and dose values are identical to the corresponding values at 100 m.
- It is assumed that the ventilation system is shut down at the time of the accident, so that no credit is taken for the reactor building filtration system.

The results included in this evaluation include:

- doses to the last staff member leaving the reactor hall after the postulated accident
- doses to a member of the public present at the site fence
- doses to the member of the public living closest to the reactor.

Dose calculations are normally done using specific information describing each site and location relative to surrounding permanent residences. Since existing MNSRs have a wide variety of local conditions and individual regulatory bodies, dose calculations were done for several exposure times and distances in an attempt to cover most situations.

Doses to the Maximum Exposed Worker

The first case evaluated for the postulated accident is that of the staff members who are present in the reactor hall during the accident. It is assumed that the last staff member evacuates the reactor hall 5 minutes after the event. This time is adequate to take the necessary actions specified in the operating procedures. Doses were also calculated for evacuation times of 10 minutes and 30 minutes. The dose rate and the activity concentrations in the air of the reactor hall during these times were based on an assumed volume method in which the radiological material is dispersed evenly in the containment/confinement volume over a specified time period (1 hour is used here). For example, to obtain the dose for 5 minutes, the dose rate in mSv/h was divided by 12 (60 minutes/12 = 5 minutes).

The effective (whole body) doses and thyroid doses that were calculated are shown in Table 6. For a typical evacuation time of 5 minutes, the effective (whole body) doses were 0.43 mSv for the HEU core and 0.49 mSv for the LEU core, in accord with the higher power level and longer lifetime of the LEU core. These doses are far less than the effective (whole body) dose limit of 50 mSv/year recommended by IAEA Safety Series No. 115. Similarly, the thyroid doses were calculated to be 0.85 mSv and 0.97 mSv for the HEU and LEU cores, respectively. These doses are far less than the thyroid dose limit of 1000 mSv/year recommended by IAEA Safety Series No. 115.

Doses to the Maximum Exposed Member of the Public

The dose for the maximum exposed member of the public was evaluated for the case that a person stands during the accident at a fence that separates the reactor site from the public area. Doses were computed at distances of 100 m, 300 m, and 500 m from the reactor. Furthermore, it is assumed that the person stays at the location for 2 hours, but cases were also calculated for 4 hours and 8 hours. After this time, the area at the public perimeter of the site is assumed to be evacuated by the security staff. No ingestion is assumed to take place during the considered time period. Doses computed for 2 hours are sufficient to consider all effects due to inhalation. The results are included in Table 6.

The effective (whole body) doses were calculated to be 1.18×10^{-3} mSv for the HEU core and 1.34×10^{-3} mSv for the LEU core assuming an exposure time of 2 hours at a distance of 100 m from the reactors, with a building leak rate of 20 volume% per hour. These doses are far less than the effective (whole body) dose limit of 1 mSv/year recommended by IAEA Safety Series No. 115. For the same conditions, thyroid doses of 3.25×10^{-3} mSv and 3.71×10^{-3} mSv were obtained for the HEU and LEU cores, respectively. These doses are far less than the thyroid dose limit of 20 mSv/year recommended by IAEA Safety Series No. 115.

Doses to the Maximum Exposed Permanent Resident

Doses were calculated for cases in which the closest permanent resident is assumed to be 300 m, 600 m, 1000 m, and 10,000 m from the reactor under the assumed conditions. The wind is assumed to blow in the direction of the permanent residence. The doses are computed for 24 hours, which is sufficient to consider all effects due to inhalation. These results are included in Table 6.

The effective (whole body) doses were calculated to be 9.28×10^{-4} mSv for the HEU core and 1.06×10^{-3} for the LEU core, assuming that the residence is located 300 m from the reactor and the building leak rate is 20 volume% per hour. These doses are far less than the effective (whole body) dose limit of 1 mSv/year recommended by IAEA Safety Series No. 115. For the same conditions thyroid doses of 2.66×10^{-3} mSv and 3.04×10^{-3} were calculated for the HEU and LEU cores, respectively. These doses are far less than the thyroid dose limit of 20 mSv/year recommended by IAEA Safety Series No. 115.

Additional Dose Calculations

The dose calculations in Table 6 incorporated an assumed building leakage rate of 20 volume% per hour. A building leakage rate of 100 volume% per hour is not considered to be credible. Nonetheless, additional dose calculations were done for representative cases with a building leakage rate of 100 volume% per hour in order to establish an upper bound on the doses that may be expected due to varying this parameter. Again, the calculated doses are far below the dose limits recommended in IAEA Safety Series No. 115. These results are included in Table 6.

Table 6. Comparison of Calculated Doses with Dose Limits Specified by IAEA Safety Series No. 115 for Key Exposed Individuals

Exposed Individual	Exposure Time	Distance	Effective (Whole Body) Dose			Thyroid Dose		
			Calculated Dose, mSv		Dose Limits, mSv	Calculated Dose, mSv		Dose Limits, mSv
			HEU	LEU	IAEA SS No. 115	HEU	LEU	IAEA SS No. 115
Assumed Building Leak Rate of 20 Volume% per Hour								
Maximum Exposed Worker	5 min	-	0.43	0.49	50 / year	0.85	0.97	1000 / year
	10 min	-	0.86	0.98		1.70	1.94	
	30 min	-	2.58	2.94		5.11	5.82	
Maximum Exposed Member of Public	2 hr	100 m	1.18x10^{-3}	1.34x10^{-3}	1 / year	3.25x10^{-3}	3.71x10^{-3}	20 / year
		300 m	5.72x10^{-4}	6.52x10^{-4}		1.58x10^{-3}	1.80x10^{-3}	
		500 m	2.37x10^{-4}	2.71x10^{-4}		6.55x10^{-4}	7.47x10^{-4}	
	4 hr	100 m	1.63x10^{-3}	1.86x10^{-3}		5.05x10^{-3}	5.76x10^{-3}	
		300 m	7.89x10^{-4}	9.02x10^{-4}		2.45x10^{-3}	2.80x10^{-3}	
		500 m	3.28x10^{-4}	3.74x10^{-4}		1.02x10^{-3}	1.16x10^{-3}	
	8 hr	100 m	1.94x10^{-3}	2.22x10^{-3}		6.83x10^{-3}	7.79x10^{-3}	
		300 m	9.43x10^{-4}	1.08x10^{-3}		3.32x10^{-3}	3.78x10^{-3}	
		500 m	3.91x10^{-4}	4.48x10^{-4}		1.38x10^{-3}	1.57x10^{-3}	
Maximum Exposed Permanent Resident	24 hr	300 m	9.28x10^{-4}	1.06x10^{-3}	1 / year	2.66x10^{-3}	3.04x10^{-3}	20 / year
		600 m	2.84x10^{-4}	3.24x10^{-4}		8.13x10^{-4}	9.28x10^{-4}	
		1000 m	1.23x10^{-4}	1.41x10^{-4}		3.53x10^{-4}	4.03x10^{-4}	
		10,000 m	4.37x10^{-6}	5.00x10^{-6}		1.25x10^{-5}	1.43x10^{-5}	
Assumed Building Leak Rate of 100 Volume% per Hour (to Establish an Upper Bound)								
Maximum Exposed Worker	5 min	-	0.43	0.49	50 / year	0.85	0.97	1000 / year
Maximum Exposed Member of Public	2 hr	100 m	3.45x10^{-3}	3.93x10^{-3}	1 / year	8.90x10^{-3}	1.02x10^{-2}	20 / year
Maximum Exposed Permanent Resident	24 hr	300 m	1.77x10^{-3}	2.02x10^{-3}	1 / year	3.95x10^{-3}	4.50x10^{-3}	20 / year

Considering the conservative approach taken for calculation of the radionuclide inventory and the assumed meteorological conditions, doses for more realistic conditions will actually be significantly lower than those that were given in Table 6.

6. Beyond Design Basis Accident / Maximum Hypothetical Accident

The Beyond Design Basis Accident (BDBA) is sometimes called the Maximum Hypothetical Accident (MHA). It is described for purposes of emergency planning and is always a postulated accident that is more severe than the DBA.

The following scenario is assumed for the BDBA [1, 2]:

- The reactor building collapses.
- The reactor vessel water and the pool water leak at a rate of 4 m^3/hr.
- The reactor core is exposed to air after 6 hours.
- The HEU reactor was operating at 30 kW and the LEU reactor was operating at 34 kW.
- The reactor has been operating for 10 years at full power, 2.5 hours a day, 5 days a week.
- The reactor scrams at the beginning of the accident sequence.

Under these conditions, the HEU or the LEU cores would be cooled by natural circulation of air and by thermal radiation. The cores would not melt. Any exposure will be external exposure from the unshielded core.

Fission product activities for unshielded generic HEU and LEU cores were calculated using the ORIGEN2 code. The power history used for the HEU core was based on a power level of 30 kW, and a power level of 34 kW for the LEU core. Table 7 compares the total activity of core fission products that were calculated for the generic HEU and LEU cores with the corresponding data in two HEU MNSR SARs [1, 2]. This comparison is quite good, given that the SARs simply state values of the total fission product activity with no justification of how the data was obtained.

Table 7. Estimated and Calculated Core Fission Product Activity with Cooling Time

	Total Activity of Fission Products in Core (TBq)		
Cooling Time	**MNSR SARs Estimated [1,2]**	**Generic HEU Core Calculated**	**Generic LEU Core Calculated**
1 min	1400	2109	2521
1 hr	170	508	611
6 hr	99	200	240
12 hr	91	146	178
1 day	80	107	132
5 days	65	61	72
10 days	58	49	56
30 days	44	32	37
60 days	-	24	27
90 days	-	19	22

Gamma dose rates from the SARs of two HEU MNSRs [1,2] at different locations for 6 hours, 1 day, and 30 days after the BDBA are shown in Table 8, along with calculated gamma dose rates for the generic LEU core. The gamma source for the LEU core is estimated to be 1.19 times higher than for the HEU core based on the 18-group spectral data shown in Table 4. The reason for this factor of 1.19 is that the LEU core has a higher power level and a longer lifetime.

Table 8. Gamma Dose Rates at Different Locations After the BDBA.

Time after Accident	Radiation Dose Rate, mSv/h				
	In the Reactor Building		Out of the Reactor Building		
	Top of Reactor (Restricted)	Reactor Hall	Balcony 10 m Away	Office20 m Away	Office 50 m Away
HEU Core [from 1, 2]					
6 hr	1.7×10^{2}	8.9×10^{-1}	1.6×10^{-2}	7.6×10^{-3}	2.7×10^{-3}
1 day	1.2×10^{2}	6.7×10^{-1}	1.1×10^{-2}	5.3×10^{-3}	1.9×10^{-3}
30 days	5.0×10^{-1}	3.3×10^{-1}	4.8×10^{-3}	2.2×10^{-3}	7.9×10^{-4}
LEU Core (estimated)					
6 hr	2.0×10^{2}	10.7×10^{-1}	1.9×10^{-2}	9.1×10^{-3}	3.2×10^{-3}
1 day	1.4×10^{2}	8.0×10^{-1}	1.3×10^{-2}	6.4×10^{-3}	2.3×10^{-3}
30 days	6.0×10^{-1}	4.0×10^{-1}	5.8×10^{-3}	2.6×10^{-3}	9.5×10^{-4}

The effective (whole body) dose limit for the maximum exposed worker recommended in IAEA Safety Series 115 is 50 mSv per year. Except for the controlled area immediately above the core and the reactor hall, all dose rates would result in doses far below the dose limit of 50 mSv per year and would permit emergency operations to proceed.

7. Conclusion

Radiological safety analysis for all of the postulated accident scenarios for conversion of MNSR reactors from HEU to LEU fuels were analyzed with the conclusion that calculated radiological doses for both cores remain within the regulatory dose limits recommended by IAEA Safety Series No. 115 with substantial margins.

Since the generic MNSR HEU and LEU cores are very close to the design of existing MNSRs, the methodology and conclusions of these analyses should be applicable to most if not all existing MNSRs.

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