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Radiological Dose Assessment for Ghana Research Reactor-1 at Shutdown Using Dispersion Model: Conversion from High-Enriched Uranium to Low-Enriched Uranium Fuel

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Ghana Research Reactor-1 is a miniature neutron source reactor (MNSR) which is currently fuelled with highly-enriched uranium (HEU) aluminium alloy fuel. Efforts are underway to convert the research reactor fuel to low-enriched uranium (LEU) oxide fuel. The project is coordinated research work funded by the International Atomic Energy Agency (IAEA) through its Coordinated Research Project (CRP) on Core Conversion. The research project was started with thermal hydraulic and neutronic calculation on both fuels. Radiological dose assessment as part of a safety assessment requirement needs to be carried out before the commencement of the core conversion project.

As such, dose assessment was credibly estimated by employing computer software (Health Physics Code HotSpot Version 3.0) developed by Lawrence Livermore National Laboratory. The code uses the Gaussian plume model for atmospheric dispersion and deposition of radionuclides based on meteorological and demographic site information. The latest IAEA guidelines for radiological dose assessment were considered in the estimation of released radionuclide doses. The anticipated estimated radionuclides released provided a comprehensive theoretical and real basis for estimating the committed equivalent dose (CED) covering the emergency and the low populated zone as expected in most severe accident scenarios. An isotope depletion analysis code ORIGEN-S coupled with MCNP5 code for neutron flux generation was used to study possible available radionuclides present in the reactor core at shutdown. Some few released radionuclides were selected from the inventory generated from the HEU core. The selected radionuclides were used in the dispersion code for dose estimation. The total activity value of two selected radionuclides (iodine and cesium) from the inventory in curies were 2.067E-03 and 6.20E-4, respectively. The values are based on the release fraction of the selected nuclides. The CED values estimated were found to be in agreement with the IAEA and US-NRC regulatory acceptable limit of 1 mSv received as public exposure and 50 mSv for radiation worker exposure in a year. The study results can be recommended when establishing the required emergency planning zones around the Ghana Research Reactor-1 facility in future.

Keywords: ORIGEN-S, HotSpot code, atmospheric dispersion, committed equivalent dose.

Introduction

The U.S. Department of Energy (DOE) and International Atomic Energy Agency (IAEA) through Atomic for Peace Initiative have agreed to convert all research reactors operating on high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. The aim of the conversion process is to eliminate the proliferation of weapons grade materials and to prevent the use of highly-enriched uranium fuel that could be used in unauthorised nuclear weapons. The projects and activities have directly supported the Reduced Enrichment for Research and Test Reactors (RERTR) programme, as well as the efforts of returning all research reactor fuel to the country where it was originally enriched. As part of U.S. nuclear non-proliferation initiatives and the RERTR programme, there has been a consensus to convert all HEU fuelled research reactors to LEU fuel. Ghana Research Reactor-1 (GHARR-1), which is a Chinese Miniature Neutron Source Reactor (MNSR), falls under reactors categorised under the Russian Research Reactor Fuel Return (RRRFR) programme. In that, the DOE and the IAEA have agreed with China to convert the Ghana Research Reactor-1 (GHARR-1) HEU core to a low enrichment core.

The Ghana Research Reactor core conversion study began in 2006 with the IAEA Coordinated Research Program (CRP), with the objective of designing an LEU core with similar operational capabilities and acceptable safety margins under both normal and accident conditions as the original HEU core. However, neutronic and thermal-hydraulic calculation of the reactor design was carried out to ascertain the viability of the whole conversion process (Akaho et al., 1995). The studies also took into account radiological dose assessment. This study was carried out before the commencement of the core conversion activity after the reactor had been shut down for almost a year. Such dose assessment requires a scale of accidents, release of possible radionuclides and site-specific meteorological conditions. It is usually vital to consider the scale of a radiological accident since some come with high occurrence probability but with less health effects, while some may have low occurrence probability but high health effects (INES, 2008). Thus, making the scale of accidents is very essential when considering radiological dose assessment.

Accuracy in estimating consequences of a radiological accident in terms of concentration and a radiological dose mainly depends on the use of a better and reliable dispersion model. Modelling of atmospheric dispersion of release radionuclides into the atmosphere is normally the first step of dose assessments (Raza et al., 2005). Not all atmospheric dispersion models are applicable to situations associated with radionuclide releases even with the improvement in atmospheric dispersion modelling computation power (IAEA-TECDOC, 1986). Dose assessment was determined by using the Health Physics code HotSpot, a state-of-the-art computer code which deploys the Gaussian plume model for atmospheric transportation, diffusion and deposition of radionuclides away from their release point within the vicinity of a release point using site-specific data related to the prevailing meteorological condition and demography. The details of models and methods included in HotSpot 3.0 are given in S. G. Homan et al. (2013).

The output results from the dispersion code were given as the committed effective dose (CED) of the two radionuclides (iodine and cesium) that are considered in this study. The accumulated CED was estimated by considering a hypothetical accident scenario. Various boundary condition assumptions of an accident scenario for radiological assessment for nuclear facilities were considered since it helps in background radiation data assessment and licensing requirements consideration. An approach which is based on the Regulatory Guide from U.S. Nuclear Regulatory Commission (USNRC) for estimating radiological releases to the environment (NRC, 2000) was adopted in this research work.

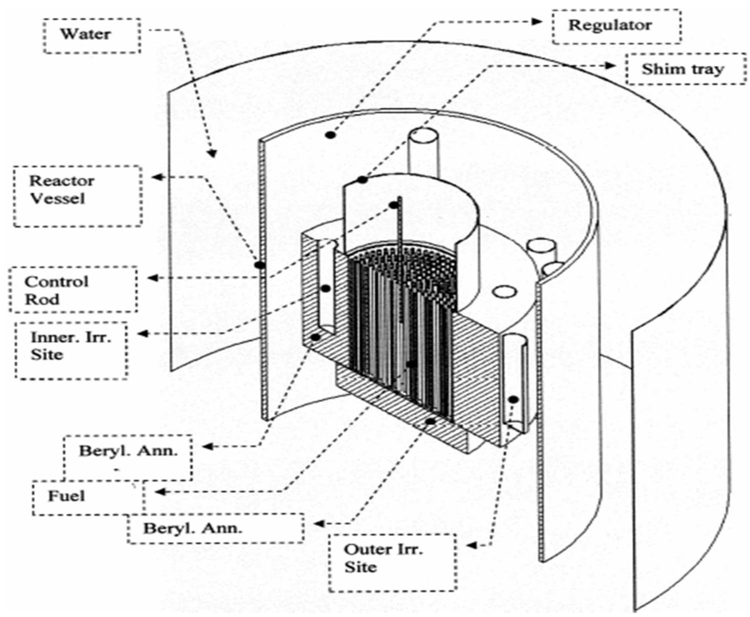
The study was limited to the use of the straight-line Gaussian plume model (GPM). The reason is that the straight-line Gaussian plume model (GPM) usually predicts estimated contaminant concentration results with very minor error margins with respect to the release point. The model is also simple and computationally inexpensive and accounts for realistic estimated results within the vicinity of the release point as it conveys pollutants instantly within the entire modelling area. The Gaussian plume model is usually used to estimate how much reduction has occurred during the transport of pollutants from a source and consequently to project the pollution concentration at the ground level (Adel et al., 2008).

Brief description of Ghana Research Reactor-1 (GHARR-1)

The Ghana Research Reactor-1 (GHARR-1) is a low power research reactor (LPRR) with the nominal power of 30 kW. It is a small, safe nuclear facility, which employs HEU as fuel, light water as a moderator, a coolant and a shield, and beryllium as a reflector. Fig. 1 shows the core configuration of Ghana MNSR.

Fig. 1

Core configuration of the GHARR-1 reactor



The reactor is cooled by natural convection. A gas purge system provides means to remove gases like argon-41 (^{41}Ar) accumulated in the empty space over the reactor. For experimentation, 10 irradiation sites are provided, 5 of the sites are located in the beryllium reflector and are called the "inner sites", while the rest that surround it are named the "outer sites". Thermal neutron flux levels of 1 x 10^{12} n/cm^2-s and 5x 1011 n/cm^2-s are available in the inner and the outer sites, respectively, at the rated power. Reactor safety is ensured by its inherent safety characteristics, which are imparted by its low excess reactivity, under-moderated core design and negative temperature coefficient of reactivity. The reactor has a self-limiting power peaking feature with a peak power approaching about 100.2 kW when all of its reactivity (~4.0 mk) is released in an uncontrolled manner (Akaho et al., 2002). The reactor can be operated either manually or automatically by the use of a microcomputer close loop control system or through a conventional control console. Table 1 shows the technical specification of the Ghana Research Reactor-1 (GHARR-1) facility.

The reactor is normally operated on demand for training of students, reactor physics experimentation and neutron activation analysis.

Theoretical description of codes

ORIGEN-S and HotSpot 3.0 were the two simulation codes employed for isotopic core inventory depletion analysis and atmospheric dispersion transport of all possible released radionuclides from the reactor, respectively.

Core decay at shutdown

180 days in 10 steps were necessary to generate the isotopic inventory for all the 344 fuel pins. The available fission products or radionuclides decay following irradiation was identified by the following conditions:

Power = 1.209E-03MW, burnup = 9.2785E+00MWD, flux= 1.95E+11 n/cm^2-s, nuclide concentrations, gram atoms basis = 0.001101 MTU.

These conditions were used to develop an input deck for the core depletion analysis to generate the output

Table 1

The main specifications of GHARR-1

Type	Tank-in-pool
1	2
Nominal core power	30 kW (th)
Coolant/Moderator	Deionised light water
Loading of U-235 in core	998,12 g
Reflector	Metallic Be alloy
Excess reactivity – cold, clean	4 mk
Daily operation fluence in inner irradiation sites	<9x10^{15} n/cm^2
Fuel life in core	>9x10^{19} n/cm^2 .s
Neutron flux at inner irradiation sites	3x10^{12} n/cm^2, stability ± 1%, horizon vertical variation <3%
Number of irradiation sites	5 inner sites at 1x10^{12} n/cm^2.s (max), 5 outer sites at 5x10^{11} n/cm^2.s (max)
Control rod	1, stainless steel clad, Cd absorber
Reactor operation	Manual and automatic
Temperature in irradiation sites	Inner sites <50 °C; outer sites <45 °C – pool temperature of 20 °C
Core reactivity temperature coefficient	-0,1 mk/°C for core temp. of 15-40 °C
Average radiation dose in reactor hall	<0,001 mSv/h

result to feed the dispersion model. The source term component in the HotSpot code for plume is accomplished by multiplying the original source term by a source-depletion factor [DF(x)]. The evaluation of this depletion factor has been described by Van de Hoven et al., 1968.

The source-depletion factor equation [eqn. 1] used in HotSpot is:

$$DF(x) = \left[\exp \int_0^x \frac{1}{\sigma_z(x) \exp\left[\frac{1}{2} \left(\frac{H}{\sigma_z(x)} \right)^2 \right]} dx \right]^{-\frac{v}{u}\sqrt{\frac{2}{\pi}}}$$

Where: *DF*(x)– Depletion factor, (dimensionless); x – Download distance (m); v – Deposition velocity (cm/s); u – Average wind speed at the effective release height, H (m/s); H – Effective release height; $\sigma_z(x)$ – Standard deviation of the air concentration distribution in the vertical direction (z axis) for either Standard terrain (adjusted for surface roughness height if applicable) or City terrain.

The dispersion transport model

The Gaussian model has been widely employed for atmospheric transport, diffusion and deposition of radioactive sources away from the vicinity and the release point. The general mathematical expression of Gaussian models implemented in the HotSpot code is given in Homann et al. (2013). The general equation [eqn. 2] calculates the steady state concentration of an air contaminant in the ambient air resulting from a point source (Macdonald et al., 2003).

$$\chi(x,y,z;H) = \frac{Q}{2\pi\mu\,\sigma_y\sigma_z}\left[\exp - \left(\frac{y^2}{2\sigma_y^{\,2}}\right)\right]$$
$$\left\{\exp\left[\frac{-(z-H)^2}{2\sigma_z^{\,2}}\right] + \exp\left[\frac{-(z+H)^2}{2\sigma_z^{\,2}}\right]\right\}$$

where: H is the height of the plume, σ_y and σ_z are respectively horizontal and vertical deviations of plume concentration distribution [*m*], Q is the uniform emission rate of pollutants [*kg/s*], x – along-wind coordinate measured in wind direction from the source [*m*], y – cross-wind coordinates direction [*m*], z – vertical coordinate measured from the ground [*m*], $\chi(x,y,z)$ - mean concentration of a diffusing substance at a point (x,y,z) [kg/m^3] and μ - mean wind velocity affecting the plume along the x – axis [*m/s*].

Materials and method

The ORIGEN-S code (Hermann et al., 1998) and the HotSpot code were employed for core depletion analysis and simulation of radionuclide concentrations and depositions, respectively.

Source term estimation

The core depletion code ORIGEN-S was employed for a detailed core analysis to estimate the available radionuclides present in the HEU. The maximum inventory at the end of life was used as the initial input for the HotSpot code for dose estimation. The release fraction from the RG 1.183 (US-NRC RG 1.183, 2000), which was initially written for PWR purposes, was used to predict the percentage of radionuclide release, as it is also a good approximation for research reactors (Raza et al., 2005). The design basic accident (DBA) scenario postulated was justified by the use of the Regulatory Guide (RG) 1.183. It was essential to develop an input deck with respect to the kind of the reactor in order to determine the isotopic core inventory during source term estimation. The nuclide composition and the continuous nuclide feed rate for radionuclides of interest were specified as a requirement for core depletion analysis.

The decay and cross section libraries were specified for the code to work with the selected data. The deck was also set up to track the photon production rate in 39 energy groups. The average burnup, flux and specific power for an irradiation were calculated. The study assumed the power history of GHARR-1 as half the power of the operational history (15kW), since that is the normal operation power. The flux spectrum of the reactor was obtained from a pre-processed cross section from MCNP5, which was used to feed the ORIGEN-S input deck.

Site-specific meteorological conditions

Site meteorological data were obtained from Ghana Meteorological Agency (GMet) database. The average sum total of the collected data was used in the dispersion code as required. The average monthly wind speed was determined for 16 directional sectors using a cup anemometer at different intervals of days. The records indicate that the maximum wind speed occurs in the month of August. The annual average wind speed of 3.1 m/s was observed. The mean daily temperature was recorded with a maximum temperature of 30.9°C. The average daily minimum temperature (hot and humid) is 24.2°C for total observations of the year. August is the coldest month with a maximum temperature of 28°C and a minimum of 22.2°C. March is the hottest with a maximum of 32.5°C and a minimum of 24.4°C. In June, the maximum average values are 29.6°C with minimum of 22.8°C. The wettest month for Accra is June with an average of 111.0 mm of precipitation falling, while the driest month is January with 7.0 mm falling.

Atmospheric condition

The atmospheric stability classes range from A, which is a very unstable condition, to F, a moderately stable condition as shown in Table 2. The table represents the distribution of Pasquill–Gifford atmospheric stability classes as an annual base of the GHARR-1 site for 5 consecutive years. These classes were estimated using a variety of meteorological measurements such as weather condition, wind speed, wind direction, precipitation, solar insulation and humidity (Homann et al., 2013). The physical characteristics of the emission source and the site-specific meteorological conditions accounted for the reason why the general Gaussian model implemented in the HotSpot code was used in this study.

The annual average wind speed of 4.10 m/s was estimated to be the predominant wind direction at 10 m above the ground level. Conservative conditions are applied when there is lack of site-specific data. However, when available, it is recommended, for dose calculations, to use site-specific information to cover the real situation (Liaw and Matos, 2012). The effective release height was calculated to be 11 m, and at that release height, the wind speed was attuned to 4.2 m/s considering the plume rise, which depends on site-specific meteorological conditions (Briggs et al., 1969). Together with other parameters, stability class C was found to be predominant justifying its use in this study. The meteorological data of the GHARR-1 site are listed in Table 3.

Atmospheric stability classification is required to quantify the dispersion capabilities of the ambient atmosphere in the air quality models for concentration predictions.

Table 2

Meteorological Pasquill–Guifford stability classes definition

Description	Stability class	Time of day	Wind speed (m/s)
1	2	3	4
Strongly unstable	A	Sunny day	< 2
Moderately unstable	B	Sunny day	3–4
Slightly unstable	C	Sunny day	4–6
Neutral	D	Cloudy/Windy	3–5
Slightly stable	E	Cloudy/Windy	2–3
Moderately stable	F	Clear Night	> 6

Table 3

Distribution of atmospheric stability classes of the GHARR-1 site from 2012 to 2016

Year	Jan	Feb	March	April	May	June	July	Aug	Sept	Oct	Nov	Dec
1	2	3	4	5	6	7	8	9	10	11	12	13
2012	B	C	C	C	B	C	C	C	C	B	B	B
2013	B	C	C	B	C	B	C	C	C	C	B	B
2014	B	A	C	B	C	B	C	C	C	C	B	B
2015	B	A	A	C	B	B	C	C	C	C	B	A
2016	A	A	B	C	B	C	C	C	B	B	A	B

Hypothetical initiating event (HIE)

A design basis accident (DBA) was selected as a maximum credible accident scenario and simulated to ascertain possible radionuclide release from the reactor vessel before the core removal process.

Postulated accident scenario

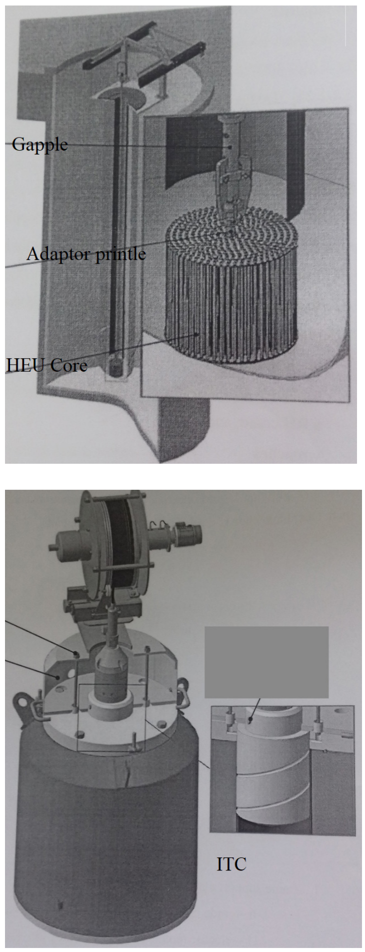
Two probable assumptions were considered:

- Stage I: After the reactor had been dormant for a period of one year, an underwater radiation resistance camera was used to inspect the reactor vessel. It was realised that some parts of the vessel were corroded assuming seepage of fission products.
- Stage II: Before the core removal activity was planned, it was decided between the operating organisation and the regulatory authority that an interim transfer cask (ITC) will be designed with a mechanical driven system (MDS) to lift the core from the reactor vessel by mounting it on top of the vessel. However, after the design completion and the interim transfer cask (ITC) mounted directly above the reactor vessel to extract or lift the core with the support of the MDS, the MDS developed a fault at a point where the core was about to enter the bottom of the ITC, and the core got stuck in mid-air (exactly between the vessel surface and the bottom of the ITC). Fig. 2 shows the design of MDS and ITC.

It was assumed that a tremendous fission product would have released into the reactor hall. The following means of leakage of radionuclides into the atmosphere were considered:

Fig. 2

Mechanical driven system (MDS) with adaptor printle to lift the core into the ITC



a a flow rate of 400 cubic meters per hour through the reactor building exhaust vent without iodine filters;

b leakage through reactor building doors and gas purge circuits.

Beyond design basis accident (BDBA), sometimes called the maximum hypothetical accident, was not expected to occur due to the nature and the inherent safety features of the reactor. The study considered it for the purposes of emergency planning and proper prediction of emergency planning zone.

Released radioactivity

Using the IAEA guidelines for post-accident condition, 10% of nuclear reactor core volatile radionuclides were assumed to be released for about 12 hours. The release was from the reactor hall, which had its exhaust ventilation system turned on as a result of human error. It was assumed that salvaging the situation by the operators was not succeeded, leading to radioactivity from the reactor core escaping through the reactor building stack in the form of gas and particulate while disseminating into the atmosphere. The release fractions applied to the calculated core activity were taken from the US-NRC regulatory guide 1.183 (US-NRC, 2000). The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 4 for PWRs.

Table 4

WR core inventory fraction released into containment

Group	Gap release phase	In-vessel phase	Total
1	2	2	4
Noble gases	0.05	0.95	1.0
Halogens	0.05	0.35	0.4
Alkaline metals	0.05	0.25	0.3
Tellurium metals	0.00	0.05	0.05
Ba, Sr	0.00	0.02	0.02
Noble metals	0.00	0.0025	0.0025
Cerium group	0.00	0.0005	0.0005
Lanthanides	0.00	0.0002	0.0002

The assumption considered led to a total possible fission products inventory in the core at 1.75 E-17 Bq, which is a summation of all the fission product estimated using the depleted analysis code.

Release height

The actual radioactivity released height may not be the physical stack height. Radioactivity rise can occur because of the velocity of stack emission, and the temperature difference between the stack effluent and the surrounding air. The rise in radioactivity results in an increase in the release height (Homann et al., 2011). The stack effluent temperature at the effective height was 30°C. The roof vent pipe has an internal diameter of 0.32 m and a stacks height of 32 m. The breathing rate was taken to be 3.33E-4 m^3 s^{-1} for an average human being under conditions of exercise (ICRP 66, 1994).

Radiation dose estimation

The volatile radionuclides of radiological significance and with a high risk of radiation health implications were analysed. The integrated source term for each radionuclide of interest in this work was then used as input for the dispersion model as shown in Table 5. The two radionuclides were selected based on their physical and chemical properties (Apostoaei et al., 1999 and Birikorang et al., 2015). They were also based on their contributions to various measures of a dose (thyroid, bone marrow, collective effective dose, etc.). Iodine-131 volatility is predominantly high among several kinds of radionuclides and its accumulation in the thyroid gland is high. Iodine can also undergo radiolysis and hydrolysis processes to produce volatile forms that can partition into the reactor building atmosphere. Under shutdown conditions, this is a particularly serious event since the reactor building is expected to be open to the surroundings and the segregated radionuclides can readily escape. Cesium-137 is a bone seeker, thus a great damage to the bone marrow, and may be present in the reactor building atmosphere as a very hygroscopic material CsOH.

The code for estimating radiological doses from the assumed hypothetical accident considered the following procedures:

a determination of fission product inventories in the core considering the operation history of the reactor;

Table 5

Selected radionuclide inventory in the HEU core at reactor shutdown

Radionuclide	Group	Inventory in curies	Release fraction	Activity released in curies
1	2	3	4	5
I-131	Halogen	2.406E-06	0.04	9.62E-7
Cs-137	Alkaline metal	2.065E-03	0.3	6.20E-4
Total		**2.067E-03**		**6.20E-4**

b radionuclide released transport behaviour from fuel into the containment building;

c the releases of fission products from the reactor building to atmosphere considering the mode of leakage pathway and filtering systems;

d the emission height, wind speed, atmospheric stability class, integrated concentration at receptor location were considered.

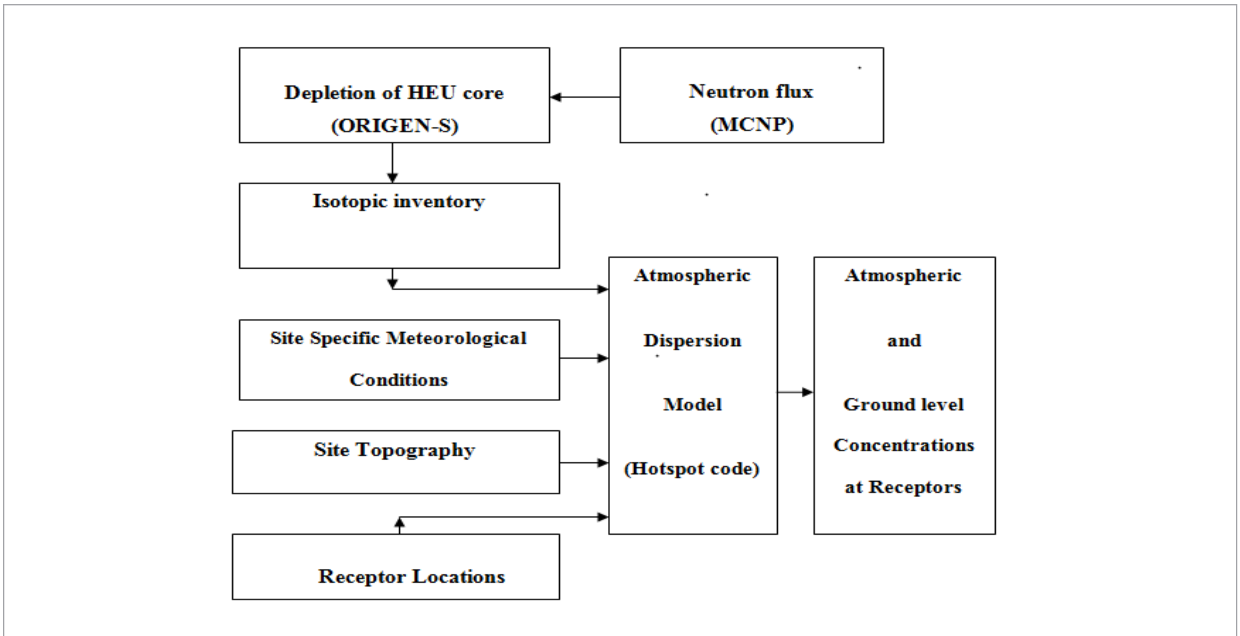
Dispersion modelling

Atmospheric dispersion was accounted for by the general Gaussian plume model in the HotSpot code, a computational tool for atmospheric transport modelling. The flow chart of the methodology adopted in this work is as shown in Fig. 3.

As fission products were released to the atmosphere, they escape from the reactor building through the vent pipe, the open door and the gas purge circuit at an air flowrate of 400 m^3h^{-1}. The fission products are immediately transported from the release point to downwind receptor locations, mostly towards the predominant wind direction at the site. The radiation dose calculations from the selected radionuclides were then estimated as final output results from the dispersion code.

Fig. 3

Flow chart of the general methodology



Result and discussion

Generated radionuclide inventory calculated values were obtained from a pre-processed cross section using the ORIGEN-S code. Most early reactor safety assessments hypothesised that severe accidents would entail the prompt release of a significant fraction of bounding radionuclide (typically iodine and cesium) to the reactor building. Several radionuclides were generated using the core depletion code but more importantly, with respect to radiological significance, the two most volatile and hazardous radionuclides (iodine-131 and cessium-137) were selected. They were selected with respect to the reactor kind understudy and the limitation of the project period. The selection of the radionuclides was also based on their radiotoxicity and the harm they can cause to human health. Their release fractions to the atmosphere were taken into consideration during the accident (US-NRC, 2000). The CED values obtained from the atmospheric dispersion code were based on the use of stability classes C, although other classes were considered in the research study, which are not accounted for in this publication. We considered only class C in this publication because it was the worst of all the conditions that were identified as shown in Table 6.

Table 6

Maximum CED values for the two selected radionuclides using stability class C

Atmospheric stability class	I-131 CED (μSv)	Cs-137 CED (μSv)	Maximum distance (km)
1	2	3	4
C	1.23E-11	1.11E-07	0.130

The results indicate that the more unstable the prevailing meteorological conditions, the higher the value of CED at a shorter distance. The results depict a decrease in the dose level when meteorological stability conditions become more stable as presented in Fig. 4.

The CED dependence on the prevailing weather condition at the site serves as a guide for operators to as and when the core should be removed. The plume rise was with respect to the downwind distance from the stack as formulated by Briggs (Briggs, 1969). The maximum CED values obtained at various distances away from the source were found to be below the acceptable limit of 1 mSv for public exposure and 50 mSv for a radiation worker (ICRP 66, 1994; ICRP 72, 1996; ICRP 60, 1991). The results confirm that even

Fig. 4

Plume centreline CED as a function of receptor downwind distance for all the stability classes

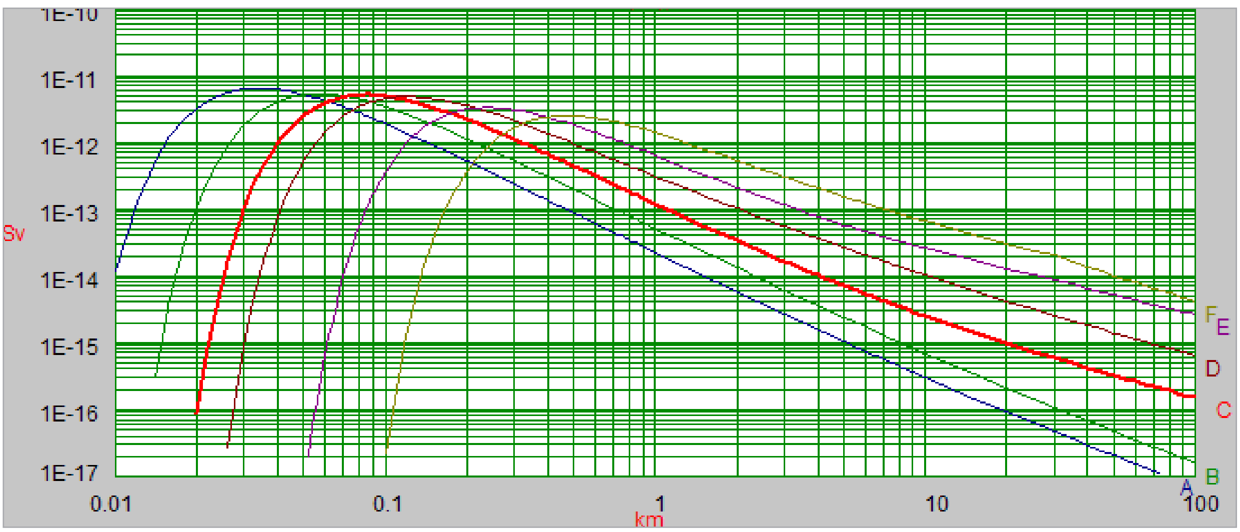
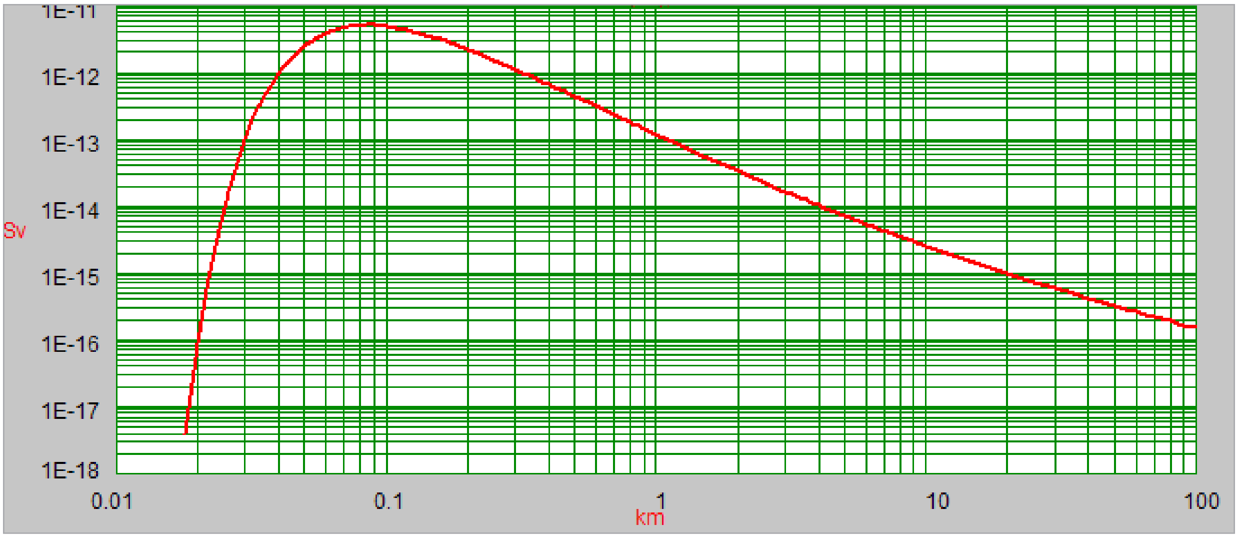


Fig. 5

Plume centreline CED as a function of the receptor downwind distance at stability class C for the I-131 isotope



under the worst accident condition, the reactor will not contribute too much radiation dose in the vicinity. More importantly, the results from the radionuclide released as considered in this study can provide an unbiased practical and theoretical basis for good approximation for emergency planning mechanism and countermeasures.

The evolution graphs of the committed effective dose as the function of the downwind distance are presented in Fig. 5 for the I-131 and Cs-137 isotopes for the predominant stability class C.

The graphical representation of I-131 isotope for stability class C is represented by Fig. 5. The graph was compared with other stability classes as shown in Fig. 4. The radiation doses rise to a certain height at a maximum distance of about 0.13 km. The dose rise comes from the release of volatized radionuclides resulting in plume buoyancy. The downward steadily decrease as shown in the graph is a result of a decrease in plume rise of radionuclides. It was due to the reactor vessel water and the pool water serving as a radiation shielding and a moderation agent. This decrease of a dose sharply occurred when the core was quickly lowered back into the vessel. Therefore, it was inferred that in order to avoid or minimise stochastic effects for a severe release there was the need to maintain both vessel and pool water to the standard level.

The dose concentration profile for I-131 and Cs-137 isotopes for ground deposition and the ground shine dose rates were estimated as the output values from the dispersion code as presented in Table 7. These values were obtained at different times as radionuclides reach designated receptor points at downwind locations.

Fig. 6 depicts a graphical representation of Cs-137 isotope distribution in the atmosphere for stability class C.

The graphical structure distribution is truly similar to that of I-131 isotope distribution layout but different in dose levels. The distribution layout was compared with all other stability classes as presented in Fig. 7.

The distribution of a target organ committed dose to some selected organs as a function of downwind was taken into account in this study as shown in Fig. 8.

The committed effective doses decrease rapidly along the path as the plume moves away from the source. This evidently conforms to the IAEA dosimetric standard of 1 mSv, which is projected to exceed a distance of 1.2 km. The estimated result shows that the skin, lung, thyroid gland, surface bone and the brain can be affected. The CED for a given exposure depends on the manner of intake and retention of various radionuclides, the possibility of concentration in the body organs and the radiological half-life of the nuclide.

Table 7

CED and other plume centreline parameters at different arrival times

Distance (Km)	CED for I-131 (μSv)	CED for Cs-137 (μSv)	Ground surface: I-131 (KBq/m^2)	Ground surface: Cs-137 (KBq/m^2)	Arrival time (hour: min)
1	2	3	4	5	6
0.100	2.6E-02	2.1E-02	8.9E+04	2.7E-01	00:01
0.200	3.5E-02	3.3E-02	2.2E+04	6.8E-02	00:03
0.300	4.2E-02	3.9E-02	1.0E+04	3.0E-02	00:05
0.400	4.6E-02	4.6E-02	5.7E+03	1.7E-02	00:07
0.500	5.1E-02	4.8E-02	3.6E+03	1.1E-02	00:09
0.600	6.0E-02	5.4E-02	2.6E+03	7.7E-03	00:11
0.700	6.5E-02	5.9E-02	1.9E+03	5.7E-03	00:13
0.800	7.2E-03	6.2E-03	1.5E+03	4.4E-03	00:13
0.900	7.7E-03	7.4E-03	1.2E+03	3.5E-03	00:17
1.000	8.5E-03	8.8E-03	9.5E+02	2.9E-03	00:019

Fig. 6

Cs-137 plume centreline committed effective dose (CED) as a function of receptor downwind distance at stability class C

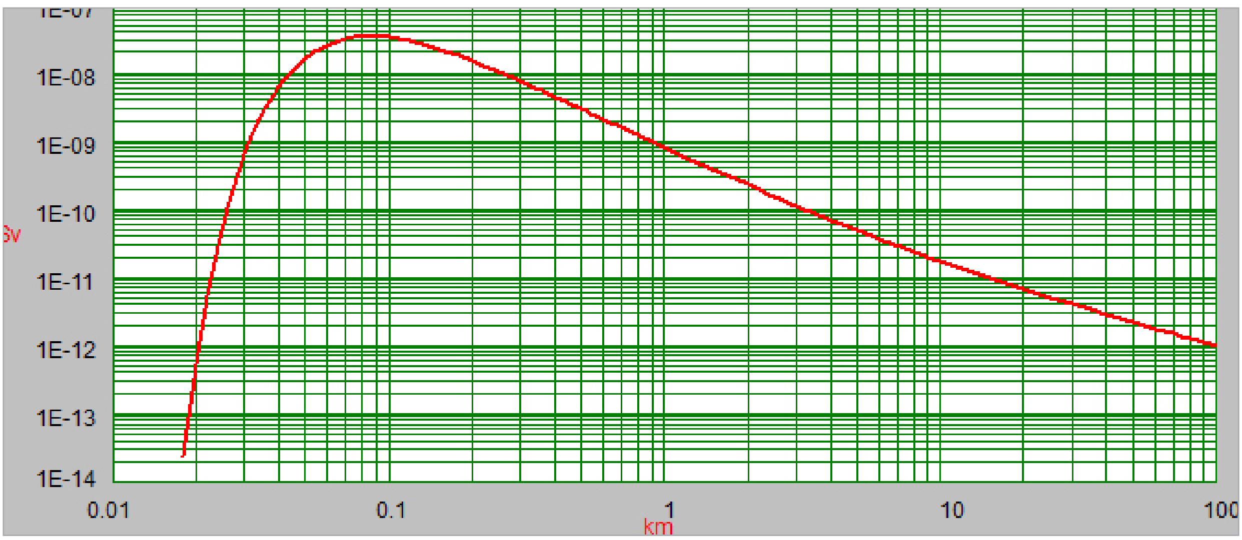


Fig. 7

Comparison of Cs-137 plume centreline CED as a function of downwind receptor locations to all other stability classes

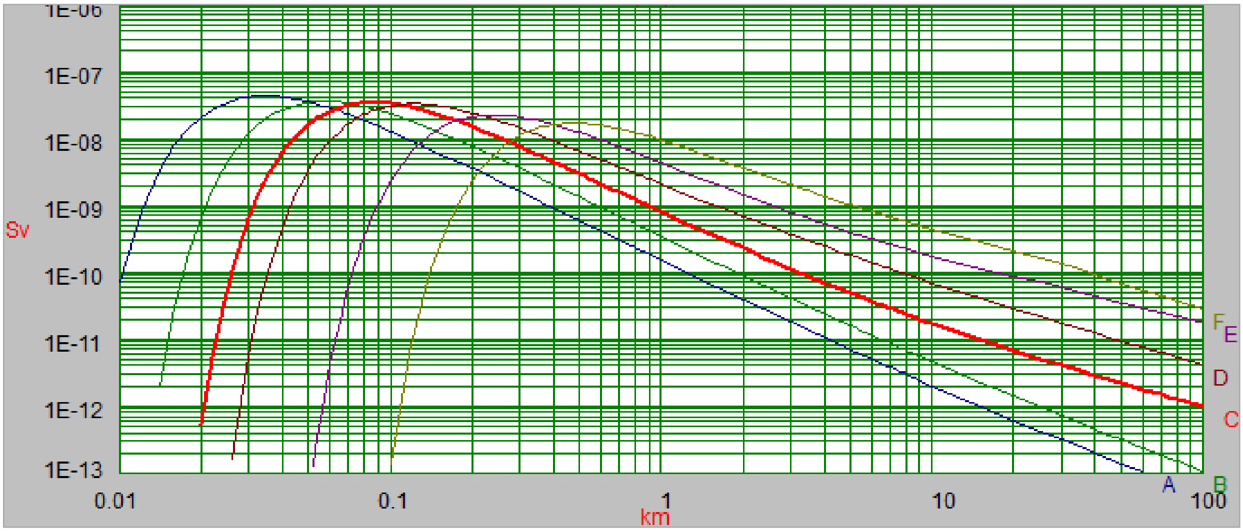
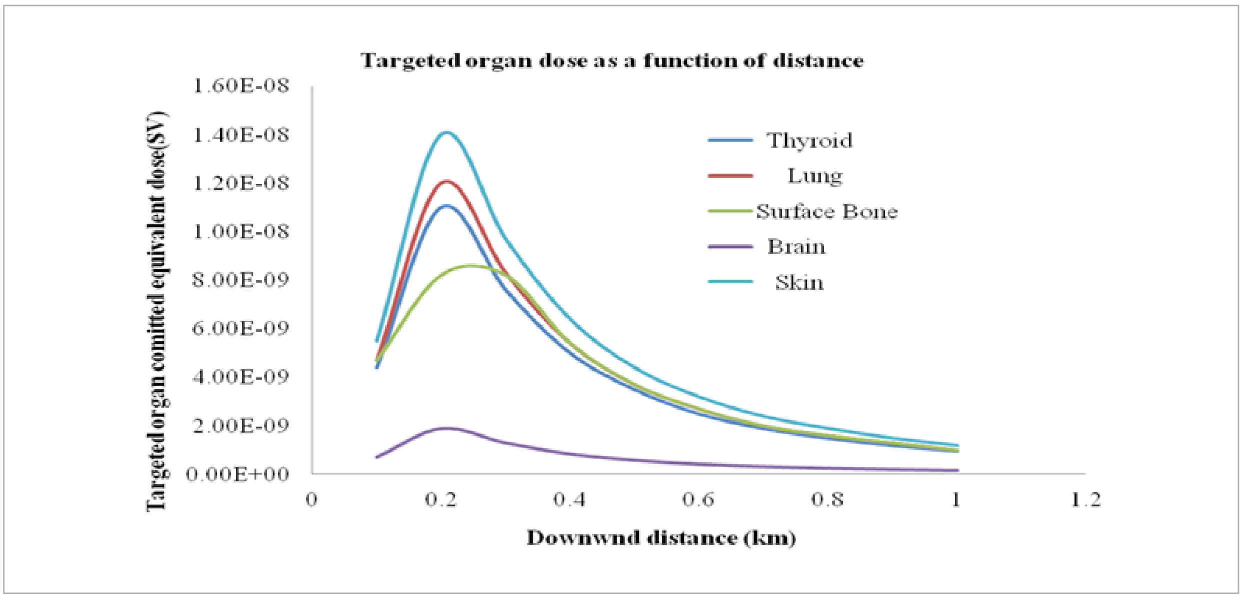


Fig. 8

Target organ committed equivalent dose as a function of the downwind location



Conclusions

Radiological safety assessment results for GHARR-1 at shutdown were successfully investigated using ORIGEN-S, a reactor physics depletion analysis code and a well-validated atmospheric dispersion code

HotSpot. The phenomenon was based on a hypothetical accident scenario where two radionuclides of radiological importance were selected and their dose level was estimated. The results were found to be within the acceptance dose limits of 1 mSv received as public exposure and 50 mSv for radiation worker exposure in a year. The study results can provide unbiased practical and theoretical bases for good approximation for a response action and an emergency planning mechanism. The margin of error due to the simulation did not significantly alter the final results. With a good degree of reliability, the hypothetical accident scenario as proposed in this research would not constitute any serious radiological effect. Conclusively, any level of radiation is dangerous to human health; hence, protective action is always beneficial.

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Ganos mokslinių tyrimų radiacinės dozės įvertinimas reaktoriui-1 naudojant dispersijos rėžimą: konversija iš labai praturtinto urano į mažai praturtintą urano kurą

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„Ghana Research Reactor-1" yra miniatiūrinis neutronų šaltinio reaktorius (MNSR), kuris šiuo metu naudojamas su aukštos kokybės prisotintojo uranu (HEU) aliuminio lydinio kuru. Šiuo metu atliekant mokslinius tyrimus kuriamas reaktoriaus kuras į mažai praturtintą urano (LEU) oksido kurą. Projektas yra koordinuotas mokslinių tyrimų darbais, finansuojamas Tarptautinės atominės energijos agentūros (TATENA) per savo koordinuojamą mokslinių tyrimų projektą (CRP) dėl pagrindinės konversijos. Tyrimo projektas buvo pradėtas su terminio hidraulinio ir neutroninio abiejų degalų skaičiavimu. Prieš pradedant pagrindinį konversijos projektą, reikia atlikti radiacinį dozavimą, kaip darbo saugos vertinimo reikalavimo dalį. Taigi, dozės vertinimas buvo įvertintas taikant Lawrence Livermore nacionalinės laboratorijos sukurtą kompiuterių programinę įrangą (Health Physics Code HotSpot Version 3.0). Kodas naudoja Gauso plumio modelį atmosferos sklaidai ir radionuklidų nusodinimui remiantis meteorologine ir demografine informacija apie svetainę. Apskaičiuojant išleidžiamas radionuklidų dozes, buvo atsižvelgta į naujausias TATENA radiologinių dozių vertinimo rekomendacijas. Išleisti numatomi apskaičiuoti radionuklidai suteikė išsamų teorinį ir realų pagrindą vertinant nustatytą lygiavertę dozę (CED), apimančią nepaprastąją padėtį ir mažai apgyvendintą zoną. Izotopų nykimo analizės kodas ORIGEN-S kartu su MCNP5 kodu neutronų srauto generavimui buvo naudojamas tyrinėti galimus radionuklidus, esančius reaktoriaus šerdyje. Keletas išleistų radionuklidų buvo atrinkti iš inventoriaus, pagaminto iš HEU šerdies. Parinkti radionuklidai buvo naudojami dispersinio kodo dozei įvertinti. Bendra dviejų išrinktų radionuklidų (jodo ir cezio) veiklos vertė iš kurių vertės buvo atitinkamai 2,067E-03 ir 6,20E-4. Vertės nustatomos pagal pasirinktų nuklidų išsiskyrimo frakciją. Apskaičiuota, kad gautos CED vertės sutampa su TATENA ir JAV-NRC reguliuojamu 1mSv ribine verte, gaunamu kaip viešai ekspozicija, ir 50mSv, kai radiacijos darbuotojai susiduria per metus. Tyrimo rezultatai gali būti rekomenduojami, kai ateityje bus nustatytos būtinos ekstremalių situacijų planavimo zonos aplink Gana tyrimo reaktoriaus-1 įrenginį.

Raktiniai žodžiai: ORIGEN-S, atmosferos dispersija, radiologinės dozės.