Methodology to Calculate Gamma Dose Contours for A Postulated Criticality Accident

R. Dranga[1], J. Wang[2]
[1] Atomic Energy of Canada Limited, Chalk River, Ontario, Canada
(drangar@aecl.ca)
[2] Atomic Energy of Canada Limited, Chalk River, Ontario, Canada
(wangjj@aecl.ca)

Abstract

Dose contour maps resulting from a postulated criticality accident need to be calculated to support emergency planning work, including establishing immediate evacuation zones and the maximum acceptable absorbed dose. This paper presents the analysis steps developed to calculate the gamma dose contour map for a postulated criticality accident. Neutron doses are not considered in this analysis since the postulated accidents are thermal systems (average lethargy of neutron causing fission for the postulated systems is between 0.033 eV and 0.082 eV), and the neutrons will be stopped by the water reflector, surrounding equipment and/or building concrete walls. The methodology is demonstrated using a conceptual facility, referred to as 'Building A', whose operations primarily involve the use of ^{235}U as the fissile material of interest. The results obtained provide useful insight into the spread of gamma radiation in a 30 m area surrounding 'Building A', and can be used for emergency planning purposes.

1. Introduction

Emergency response planning is required by CNSC regulations for any facility where a criticality accident alarm system is in use [1] [2]. The main purpose of emergency response planning in this case is to minimize the risk to personnel during an emergency response to a nuclear criticality accident outside of a reactor. It involves, amongst others, establishing the immediate evacuation zone boundary and the maximum acceptable value for the absorbed dose at that boundary. However, in order to accomplish these tasks, potential criticality accident locations need to be identified, criticality accidents must be postulated and a dose contour map, accounting for any significant shielding, must be calculated around the facility's perimeter. It is important to note that the criticality accidents postulated for emergency response planning represent conceivable sequences of events which result in an inadvertent critical assembly. Nevertheless, the postulated criticality accident is considered to be an "incredible abnormal condition"[1], with respect to the criticality safety of the facility.

This paper summarizes the methodology developed to calculate the gamma dose contours based on a postulated criticality accident. The methodology is demonstrated using postulated criticality accidents for a conceptual facility performing operations using fissile material, in particular, ^{235}U. This facility will be referred to as 'Building A' throughout this paper. The resulting gamma dose contour maps

[1] In criticality safety, an "incredible abnormal condition" is a condition which has a probability of occurrence of less than 10^{-6} per year [1].

provide the supporting information required to establish immediate evacuation zones, in compliance with CNSC regulatory requirements [1].

2. General Description and Methodology

2.1 Computer Code Package

MCNP5 [3] has been used to model the layout for 'Building A', large pieces of equipment which were deemed to provide significant gamma shielding, and a 30 m perimeter surrounding the facility. MCNP5 version 1.40 was used on a computer cluster which runs RedHat Linux version 5.5, and has 240 identical processors (10 Dual CPU Blades, Each CPU has Dual Hex Core Processors). Each model was executed using 36 CPUs.

2.2 Methods

In order to establish the correct dose contours based on the postulated criticality accident(s), the following main steps were established and implemented:

a) Created a computer model of the facility, based on engineering drawings, physical measurements, and conservative assumptions and simplifications. The model included walls, floors and ceiling, doors, windows and any other openings, pieces of equipment which were considered to provide significant shielding (e.g., large metal equipment). Small pieces of equipment, office furniture and filling cabinets did not have to be modelled as they provide only negligible shielding. A sample model of 'Building A' is illustrated in Figure 1.
b) Identified the potential locations for postulated criticality accidents, based on current facility operations (see Figure 1). This step was particularly important since all potential locations had to be identified in order to produce a complete and adequate contour map.
c) Presented a description of the postulated criticality accident for each location described in step (b). In general, this step should include a discussion on the conceivable events which must occur to result in a criticality accident and the supporting analysis showing that these steps result in a critical system (i.e., one which results in an effective neutron multiplication factor, k_{eff} of 1.000 or greater). For the purpose of this paper, a brief description of the two postulated accidents identified for 'Building A' is provided in Section 2.2.1.
d) Discussed and reported the total fission yield for the postulated criticality accident(s), in compliance with CNSC Guidance Document GD-327 [2]. The fission yield used for the 'Building A' example is discussed in Section 2.2.2.
e) Calculated using a Monte Carlo code (i.e., MCNP5) the dose contours based on each individual postulated accident, and combined the results by choosing the highest calculated dose at each tally location, around the outer perimeter of the facility. For 'Building A', the postulated accidents were represented by thermal systems (EALF[2] values between 0.033 eV and 0.082 eV), since the fissile material was homogeneously mixed with the optimal[3] amount of light water. Hence, the neutron doses were not calculated, because any neutrons escaping the

[2] EALF is the average lethargy of neutrons causing fission. Neutron lethargy is a dimensionless logarithm of the ratio of the energy of source neutrons to the energy of neutrons after a collision. This values is an indication of how thermal the system is.

[3] An optimally light-water moderated system refers to a system containing the optimal amount of light water (i.e., the amount of water which, for the most reactive geometrical configuration, results in the minimum critical mass of fissile material).

systems would be thermal neutrons and they would be stopped by the 15 cm water reflector, the concrete building walls and/or the surrounding equipment. For this reason, the example presented in this paper will only focus on gamma doses. Nevertheless, depending on the facility and the type of operations performed, neutron dose contours might also be required.

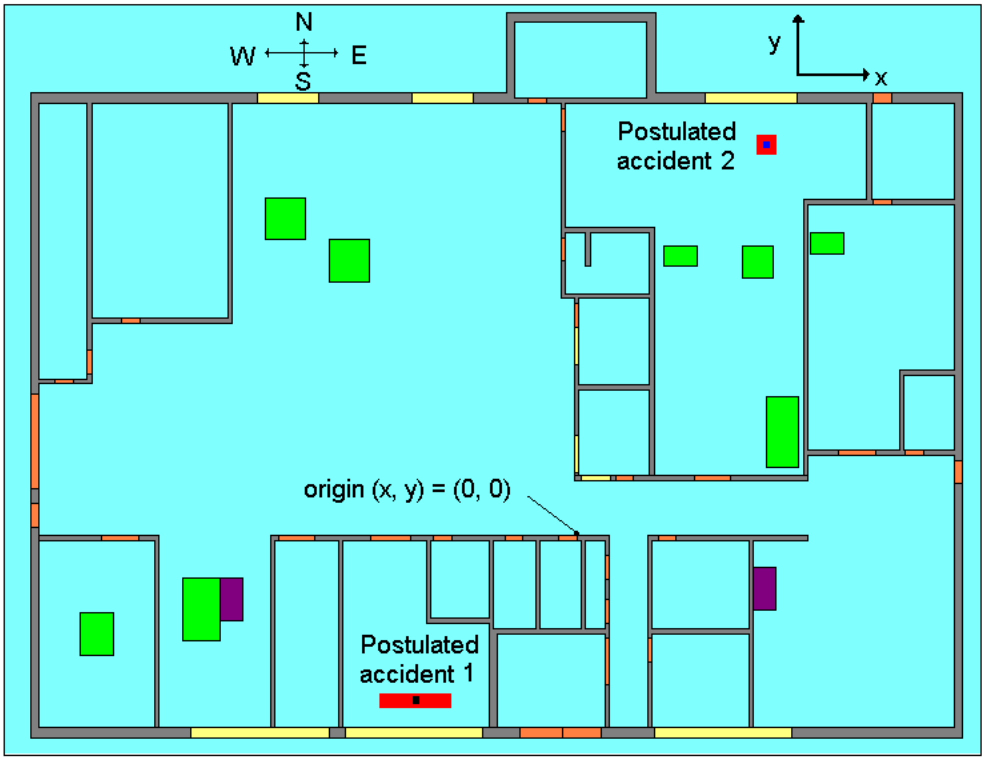

Figure 1 Computer model (X-Y view) of 'Building A' including location of the two postulated accidents and major pieces of equipment

2.2.1 Postulated criticality accidents

Two criticality accidents were postulated for 'Building A': Accident 1 involving optimally light water moderated targets composed of 21 wt% U(93.5%) and 79 wt% Al, and Accident 2 involving a homogeneous mixture of uranium with an enrichment of 93.5 wt% (i.e., U(93.5%)) and light water (see Figure 1 for the accident locations). Both postulated accidents are considered to be non-credible, yet they represent the conceivable sequence of events which would result in an inadvertent critical system and should be analyzed for emergency response planning and on-site personnel dose mitigation purposes. The criticality sources representing the postulated accidents were defined in MCNP using the KCODE card.

2.2.2 Estimated Fission Yield for the Postulated Criticality Accidents

The two postulated accidents discussed above result in a total moderator volume of 12.80 L for Accident 1 and 12.95 L for Accident 2. Using Equation 1 below [4], a total fission yield of 7.7×10^{17} fissions and 7.8×10^{17} fissions were estimated for postulated criticality Accident 1 and 2, respectively.

$$F = 6.0 \times 10^{13} \times V_s \qquad \text{(Equation 1)}$$

where, F is the total fission yield (in # of fissions) and V_s is the solution volume (in cm^3).

Furthermore, the *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, NUREG/CR-6410 [5] provides further guidelines in obtaining a bounding estimate of the total fission yield for various types of systems. The two scenarios discussed here fall under the category of small liquid systems involving less than 300 L of moderator, which would result in a total of 1.0×10^{18} fissions for the duration of the entire accident, which for the purpose of this analysis is assumed to be 1 minute. The moderator volumes characterizing the postulated accidents for 'Building A' are much lower than the upper limit of the category described in NUREG/CR-6410 [5]. Therefore, assuming a total of 1.0×10^{18} fissions/min for both postulated criticality accidents was conservative and bounding.

This value was used to normalize the MCNP results and calculate the gamma dose rates (over a period of one minute) at various locations. It is important to note that the dose rates were calculated from the total dose being incurred over a period of one minute. This was based on criticality accident design calculations and does not represent the mean dose rate for longer intervals of continuous exposure.

2.2.3 Calculating Gamma Doses Outside of 'Building A' for Dose Contour Mapping

The F6 tally in MCNP5 was used to calculate the energy deposition, in MeV/g, averaged over a volume. This tally was used throughout the models to estimate the photon doses in MeV/g at various locations around the outside perimeter of 'Building A'. The tallied volumes were modelled as elliptic cylinders (i.e., cylinders with an elliptical, rather than circular, cross-section), centered at z = 105 cm above the floor level, and the length coinciding with the length and width of the facility (see Figure 2 for an illustration of these tally volumes). The semimajor axis (i.e., the major radius, R) was 70 cm and the semiminor axis (i.e., the minor radius, r) for the ellipse was 15 cm (see Figure 3). This tally volume geometry was chosen to resemble, as closely as possible, to a human being, with the focus being on the human's trunk, which would be exposed to the highest gamma radiation. The tally volumes on the North and South sides of the facility were divided into 100 cm long sections, and photon doses were calculated for each section. The tallies on the East and the West sides were divided into 100 cm long sections between the North and South sides of the outer facility walls. Table 1 below summarizes the location of the tally volumes, with respect to the outer facility walls.

Table 1 Distance Away from Outer Facility Walls, Representing the Center of Elliptic Cylinders used for Tallying in the MCNP Models

Tally	West Side (m)	East Side (m)	North Side* (m)	South Side (m)
1	0.5	0.5	0.5	0.5
2	1.5	1.5	1.5	1.5
3	3.0	3.0	3.3	3.0
4	4.5	4.5	4.5	4.5
5	6.0	6.0	6.0	6.0
6	8.0	8.0	8.0	8.0
7	10.0	10.0	10.0	10.0
8	12.0	12.0	12.0	12.0
9	15.0	15.0	14.0	15.0
10	18.0	18.0	17.0	18.0
11	21.0	21.0	20.0	21.0
12	24.0	24.0	23.0	24.0
13	27.0	27.0	26.0	27.0
14	30.0	30.0	29.0	30.0
15	--	--	32.0	--

* These distances are slightly different than those on the South, East and West sides due to one room which is not in-line with the rest of the building (see Figure 1).

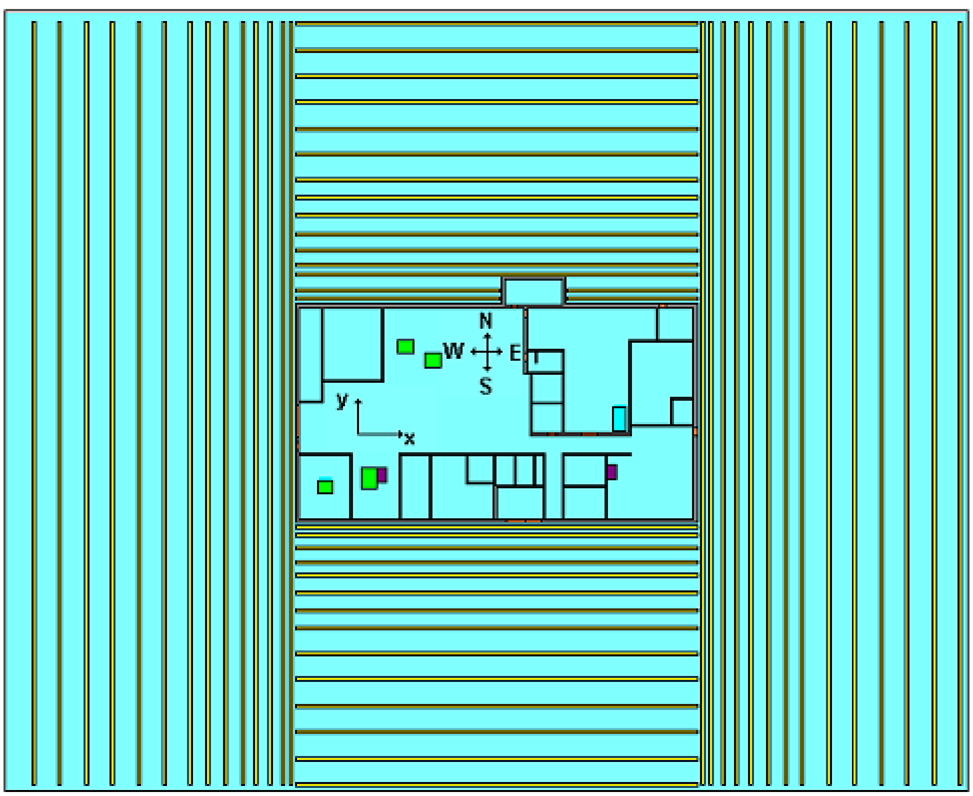

Figure 2 X-Y view (Z = 105 cm) of 'Building A' with the tally volumes modelled up to 30 m around the outer building walls.

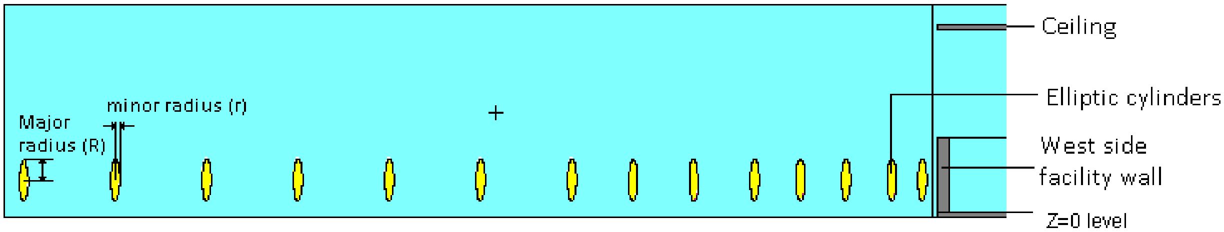

Figure 3 Y-Z view of 'Building A' (west side), with the tally volumes modelled up to 30 m away from the outer building wall.

The gamma tally results account for both fission gammas produced by the critical assembly, as well as prompt gammas produced from (n,γ) reactions between the fission neutrons and the concrete walls or the large pieces of equipment. The tally results for each segment were converted to Gy/fission and normalized to the fission yield of 1.0×10^{18} fissions/min, using Equation 2.

$$\gamma \text{ dose}\left(\frac{\text{rad}}{\text{min}}\right) = \text{Energy deposited}\left(\frac{\frac{\text{MeV}}{\text{g}}}{\text{neutron}}\right) \times 10^{6}\left(\frac{\text{eV}}{\text{MeV}}\right) \times 10^{3}\left(\frac{\text{g}}{\text{kg}}\right) \times (1.6 \times 10^{-19})\left(\frac{\text{J}}{\text{eV}}\right)$$
$$\times \text{ nubar}\left(\frac{\text{neutrons}}{\text{fission}}\right) \times (1 \times 10^{18})\left(\frac{\text{fissions}}{\text{min}}\right) \times 100\left(\frac{\text{rad}}{\text{Gy}}\right) \qquad \text{(Equation 2)}$$

The calculated nubar value for the fissile material systems discussed in this analysis is 2.437. This value was used in Equation 2 to calculate the total gamma dose in rad received over a period of one minute. The "energy deposited" term is the energy deposited in the elliptic cylinder segment (modelled as water with a density of 0.001 g/cm^3). Also recall that 1 J/kg = 1 Gy = 100 rad and 1 eV = 1.602×10^{-19} J.

3. Dose Contour Results and Discussion

3.1 Dose Contour Resulting from Postulated Accident 1

MCNP5 transport calculations using postulated Accident 1 as the criticality source were performed, and the tally results were used to create the gamma dose plot illustrated in Figure 4. As expected, the results show that the maximum dose obtained is directly in front of the window where the postulated critical assembly is located. This maximum dose is 772 rad (at 0.5 m away from the South facility wall), decreasing rapidly afterwards.

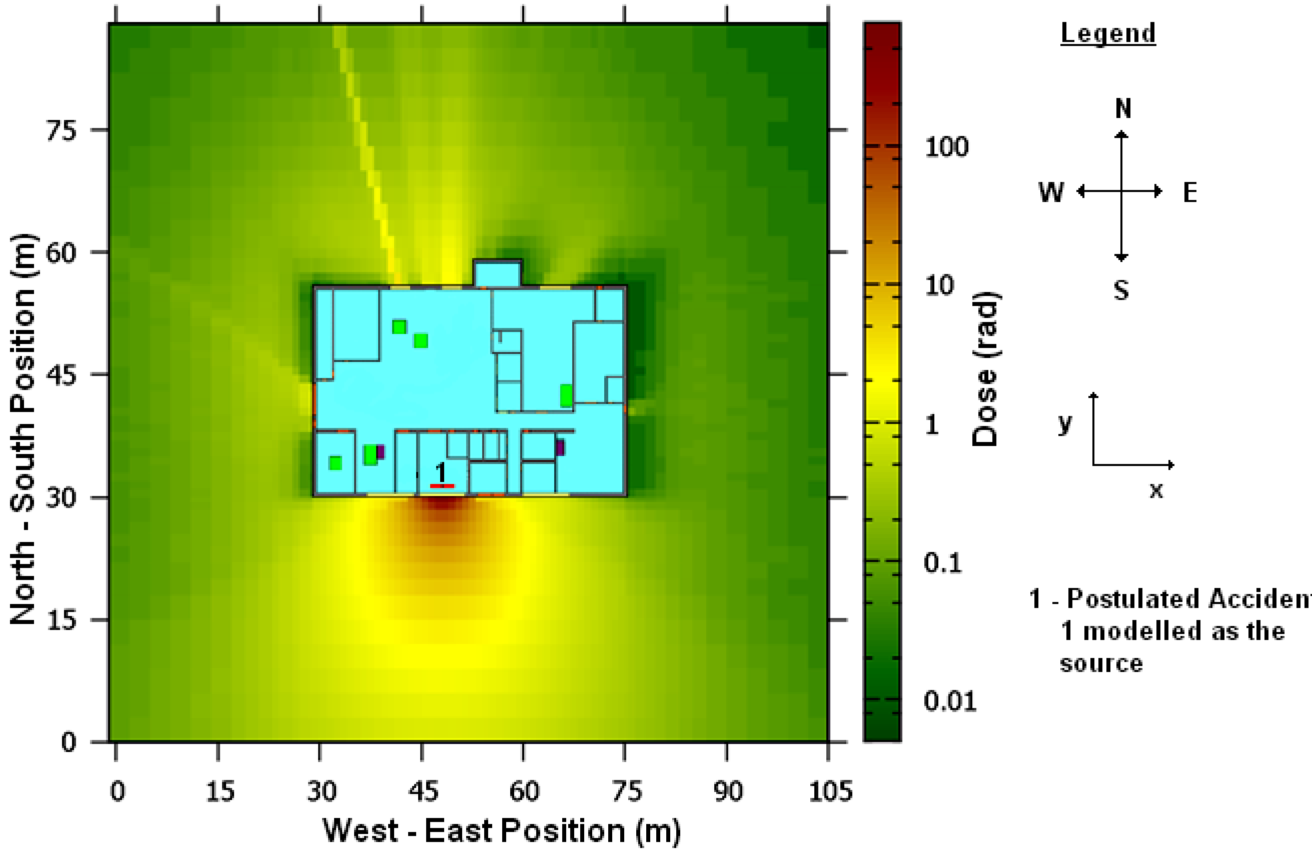

Figure 4 Gamma doses in rad resulting from postulated Accident 1

This accident was postulated to occur at a height of 100 cm above the floor level. The window, which is located south of the critical assembly, starts at 122 cm above the floor level. Therefore, the extra 22 cm of concrete provide additional shielding, helping decrease the dose to values below 20 rad at distances further than 6 m away from the facility wall. At 30 m away from the facility, the maximum dose for the west, east, and south sides are 0.06 rad, 0.09 rad, and 0.99 rad, respectively. The maximum dose calculated at 32 m away from the north facility wall is 0.31 rad. Another important observation which can be drawn based on Figure 4 is the distinctive "ray" of higher gamma doses in front of the door on the west side of the facility, and the windows located on the north side.

3.2 Dose Contour Resulting from Postulated Accident 2

The tally results for transport calculations using postulated Accident 2 as the criticality source were also obtained and are illustrated in Figure 5. The results show that the maximum dose is obtained in the area in front of the window where the criticality accident was located. Postulated Accident 2 was assumed to occur at a height of 125 cm above the floor level. The window, which is located directly in the line-of-sight of the accident starts at 122 cm off the floor level. Therefore, there is no shielding present and the gamma radiation was observed to travel a further distance through air than if shielding (e.g., concrete walls) were present. The maximum gamma dose obtained based on this postulated accident is 504 rad, at 0.5 m away from the outer north wall of the facility, directly in front of the critical assembly. As expected, this dose decreases with distance; however, even at 12 m away from the outer north wall of the facility, the dose is still above 20 rad, decreasing below this value only at distances greater than 14 m. At 30 m away from the facility wall, the maximum doses on the west, east, and south sides are 0.15 rad, 0.16 rad, and 0.19 rad respectively. At 32 m away from the North facility wall, the maximum calculated dose is 4.14 rad.

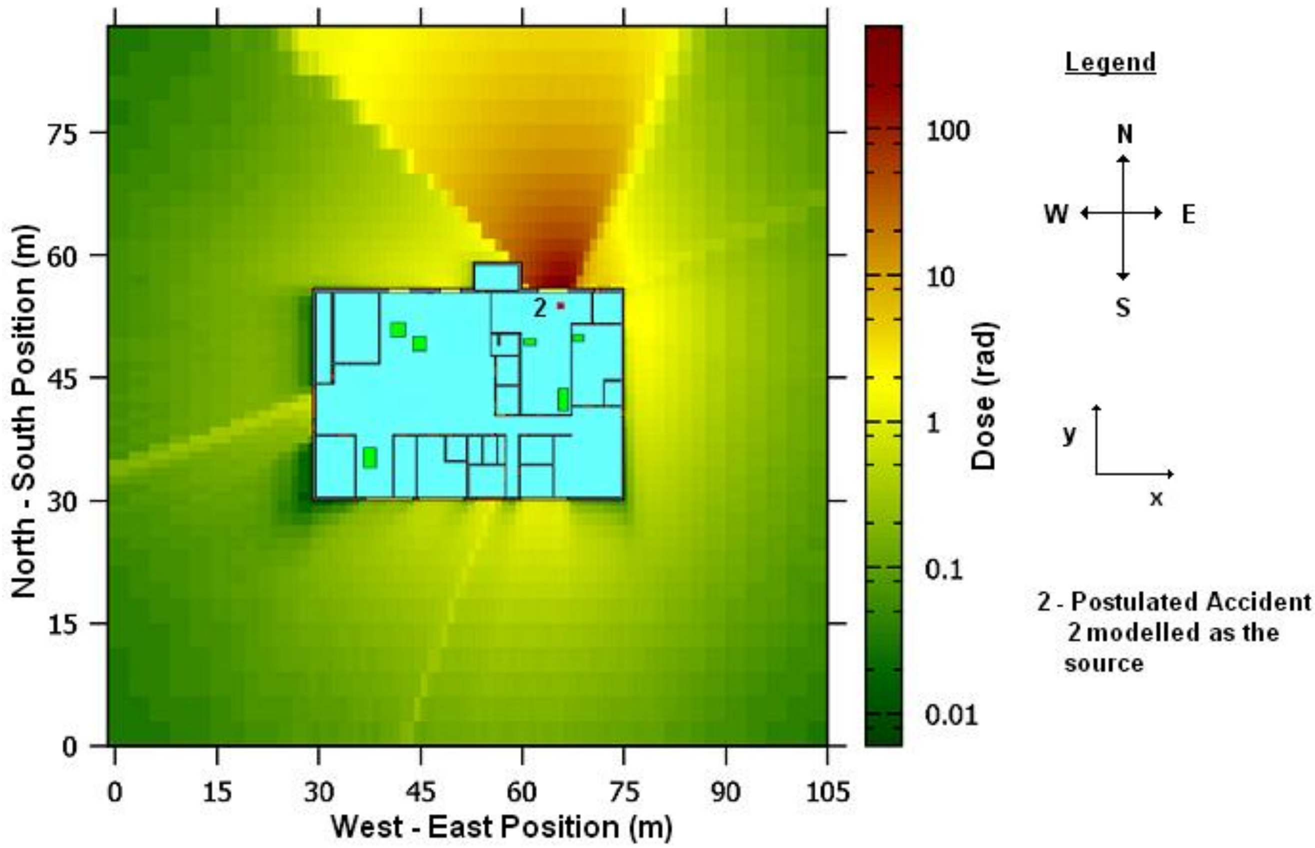

Figure 5 Gamma doses in rad resulting from postulated Accident 2

As noted in the analysis for postulated Accident 1, one can see again a distinctive "ray" of higher gamma doses in front of the door on the west side of the facility, and the doors and windows on the south side of the facility. This is expected, especially since there are fewer walls to act as shielding between these locations and the criticality source. Furthermore, the windows and the doors provide negligible shielding, in comparison to the 40 cm concrete outer facility walls.

Another interesting observation which needs to be mentioned is that at 0.5 m and 1.5 m away from the north-west, south-west and south-east walls of the facility, the doses are lower (illustrated by the dark green colour in Figure 5). This result is expected since the concrete walls (40 cm thick) provide good shielding for the gamma radiation originating from the source. However, the gamma dose will increase slightly past the 1.5 m line since gammas will be reflected off the 16 cm thick concrete ceiling (which is located at 6.0 m above the floor level).

3.3 Combined Dose Contours and Associated Relative Uncertainties

A combined and final dose contour map was produced based on the highest dose calculated at each tallied location from both postulated accidents. This combined dose map, illustrated in Figure 6, accounts for all scenarios, considers the most conservative results, and provides useful information to establish immediate evacuation zones.

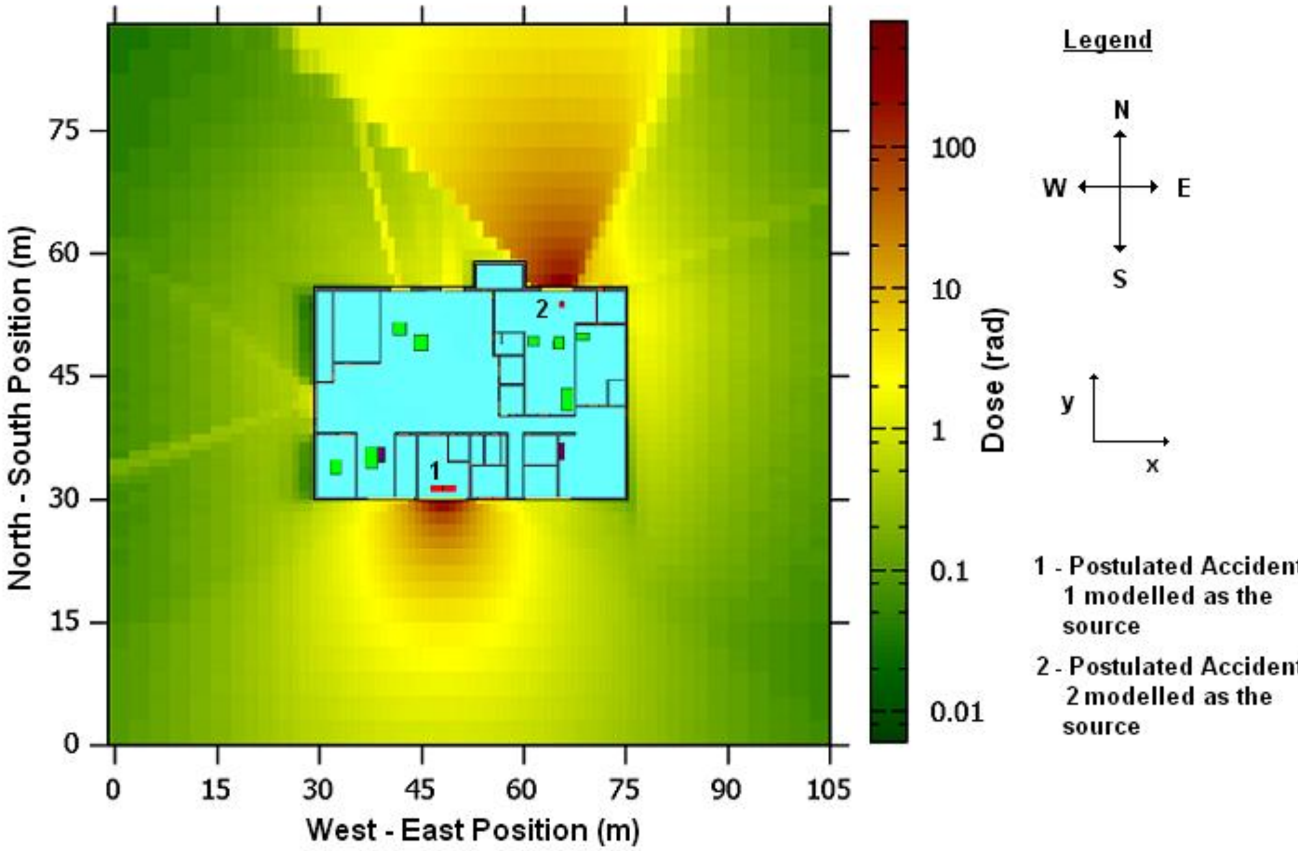

Figure 6 Combined gamma doses in rad resulting from the two postulated accidents

The “hot spots” around ‘Building A’ are, as expected, in front of the north and the south windows, where the two postulated accidents are located. However, the “hot spot” on the north side of the building extends for a much greater distance from the outer facility wall, as discussed in Section 3.2. Also, the “rays” of higher gamma doses originating from the two postulated accidents combined are still easily noticeable in front of the door on the west side and the windows on the north side of the facility. However, this effect was lost for the windows located on the south side of the facility, since the radiation doses due to postulated Accident 1 are dominant.

The relative uncertainties[4] associated with this combined dose map are illustrated in Figure 7. The highest relative uncertainty of 13.8% is obtained for the segment located 43.9 m away from the north-west corner of ‘Building A’. The dose for this segment is only 0.03 rad; hence, the higher relative uncertainty is acceptable. Moreover, it is important to note that in areas of interest, where the doses are higher, the relative uncertainties are below 2%. In the “hot spots” surrounding the windows beside which the accidents are postulated, the relative uncertainties are below 1%.

[4] Relative uncertainty values are presented as a % of the MCNP calculated tally result. In MCNP, these values represent the statistical precision as a fractional result with respect to the estimated mean value.

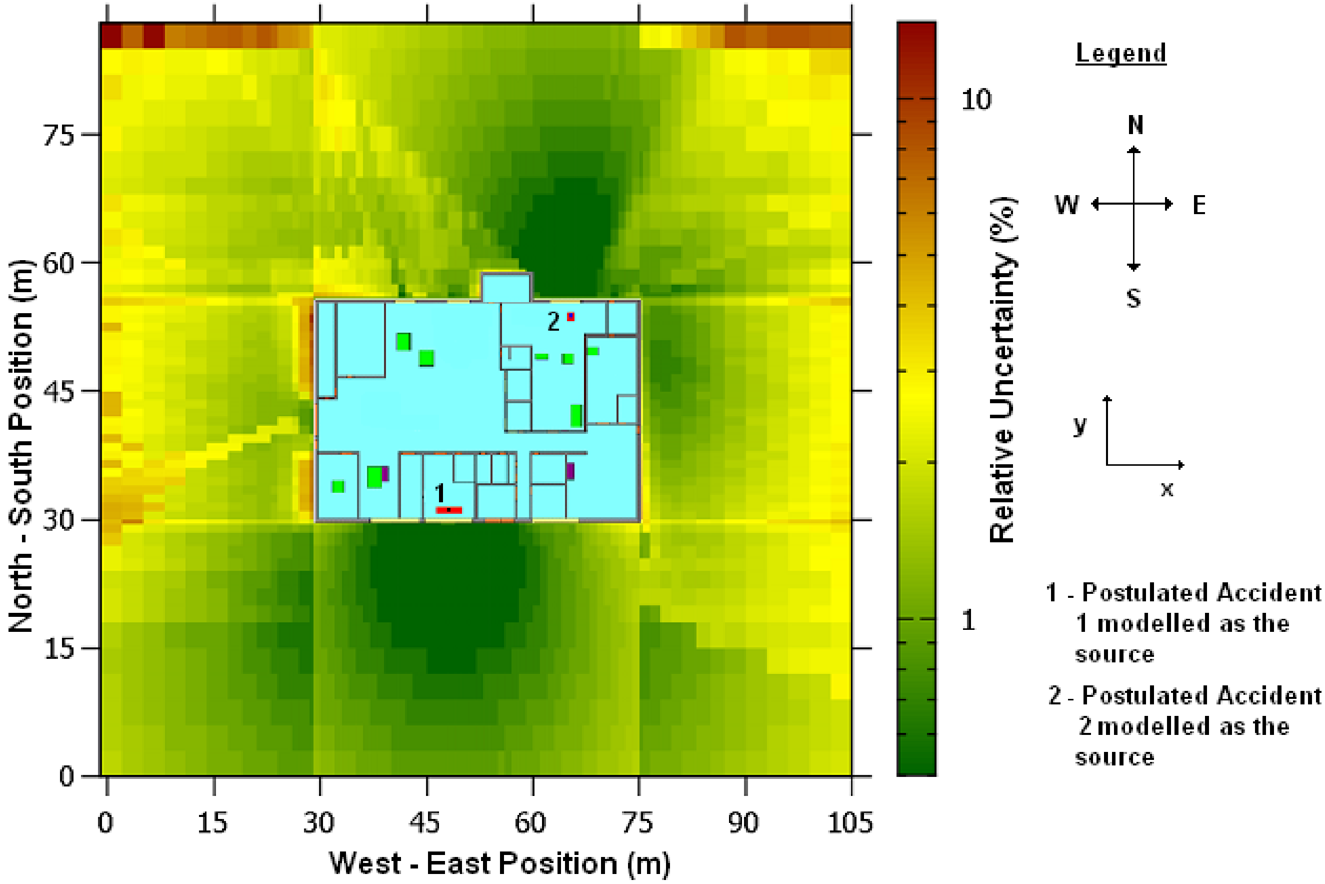

Figure 7 Relative uncertainties in the doses calculated for the contour map combining results from both postulated accidents.

4. Conclusions

The methodology designed and implemented to calculate dose contour maps based on postulated criticality accidents is summarized in this paper. The step-by-step process is demonstrated using ‘Building A’ as an example. This is a facility where processes involving ^{235}U take place. The analysis process includes the following main steps: 1) designing a computer model of the building layout, including any equipment which provides significant shielding; 2) identifying the potential location for postulated criticality accidents; 3) describing the postulated accidents, which represent a series of conceivable steps resulting in a critical assembly; 4) reporting the total fission yield for the postulated criticality accidents; and 5) calculating the dose contours for individual postulated accidents and combining the results in one dose contour map.

The methodology presented in this paper can be used to support emergency planning work and establish immediate evacuation zones for facilities at Chalk River Laboratories where criticality accident alarm detectors are in place.

5. References

[1] CNSC Regulatory Document, "Nuclear criticality safety", RD-327, Edition 1.0, December 2010.

[2] CNSC Guidance Document, "Guidance for nuclear criticality safety", GD-327, Edition 1.0, December 2010.

[3] X-5 Monte Carlo Team; “MCNP – A general Monte Carlo N-Particle transport code, version 5”, LA-UR-03-1987, Los Alamos National Laboratory, 2003.

[4] Y. Nomura, H. Okuno, “Simplified evaluation models for total fission number in a criticality accident”, Nuclear Technology, Vol. 109, pp. 142-152, 1995 January.

[5] U.S. Nuclear Regulatory Commission, “Nuclear fuel cycle facility accident analysis handbook”, NUREG/CR-6410, 1998 March.