SOURCE TERM AND RADIOLOGICAL CONSEQUENCE EVALUATION FOR NUCLEAR ACCIDENTS USING A "HAND TYPE" METHODOLOGY

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Abstract: In many cases, results from computerized accident consequences assessment models may be delayed due to the equipment malfunction or the time required to develop minimal input files and perform the calculations (typically more than five minutes). A simple nomogram (developed using computerized dispersion model calculations) can provide dispersion and dose estimates within a minute. The paper present the methodology used for these "hand type" calculation and the nomograms, figures and tables used to evaluate the dose to an individual close to the release point. In order to illustrate the use of methodology, a hypothetical severe accident scenario involving 14-MW INR-TRIGA research reactor was considered.
Key words: emergency response plan, countermeasures, health effects, nuclear accident, early containment failure

1.   METHODOLOGY

In the last decades, hand type calculations have been replaced by computerized solutions, which are much more accurate, but, preparation of an input to run the code can be a time consuming process and can require a laborious work. This is why, a place for hand calculation based on nomograms still exist in some areas. An example is emergency response to an accidental release of radioactive contaminants when the health of persons close to the accident site might be at risk.

Source term and radiological consequences analysis of research reactor accidents follows the following sequence:

1. Define the accident scenarios. Determine which scenarios are Design Basis Accidents, and which are Beyond DBA events. Select scenarios that are credible for analysis.
2. Define the radioisotope content of the reactor core for burnup conditions which *match and bound* the scenarios selected.
3. For each selected scenario, calculate the reactor time-evolution history (power and energy released vs. time, peak fuel, clad, and coolant temperatures attained, and any other needed safety parameters such as coolant pressure peak, safety system response, operator response).
4. Determine the kind and extend of fuel damage, so as to permit bounding the amount of fission products, actinides, and other radioisotopes, which are released, to the environment.
5. If there is fuel damage and radiation is released from the fuel, determine the release pathways and amounts to various key points. For example:

      a. What fraction of radioisotopes is released from the fuel?
      b. What fraction of radioisotopes remains in the coolant?
      c. What fraction of radioisotopes emerges from the coolant system into the reactor building air?
      d. What are the radiation dose rates to operations staff or to experimenters in the reactor building?
      e. What are the release pathways and rates from the reactor building?
      f. Is radiation released from the reactor building as a puff, a series of puffs, or continuously?
      g. For several wind conditions, determine the radiation dose rates downwind at the site boundary or at any other locations required by the licensing authority.
      h. Using personnel occupancy estimates, determine doses to reactor building occupants and to the general public.

6. Review the consequences analysis for all scenarios. Rank them in order of seriousness. Verify that the scenarios having the most serious consequences are both credible and well-defined. Refine the scenarios if necessary.

2. ACCIDENT SCENARIO

In order to illustrate the use of methodology in source term and radiological consequence evaluation, a hypothetical severe accident scenario involving 14-MW INR-TRIGA research reactor is considered. Thus a large part of the reactor hall roof or a heavy object escaped from the crane hook is dropped over the 14-MW TRIGA-SSR core, resulting in mechanical damage of the core. It is assumed, also, that no core melting is occurring, but only fuel-cladding rupture being involved for several 25-pins fuel bundles. In fact this is an extension of one from DBA accidents.

It is assumed, also, that no core melting is occurring, but only fuel-cladding rupture being involved for several 25-pins fuel bundles. The affected fraction of the core is 45%. Izotopic mixture of the released effluents during a reactor accident, will strongly depends on the mechanism involved in fuel damage, on the status of the safety barriers and the dinamic of the accident. The released fraction is function of the isotopes volatibility and the temperature reached by core during the accident. If the fuel cladding rupture occurs during the reactor normal operation, in main coolant loop are released: noble gases (Xe and Kr), extremely volatile fission products (iodine) and volatile fission products (Cs, Te, Ru). As the temperature rise, other isotopes could be released, also, into the main coolant loop.

In this scenario, the fuel temperature during the accident is far from melting point. For the release, we consider an experimentally determined fraction of volatile products released from the fuel material, which will be about 6.3E-04. We also consider that there is no retention of volatile fission products in the fuel-moderator material. From the fraction of damaged core, 100% of the noble gases in the fuel-clad gap are released from the fuel bundles and, subsequently, are transferred directly to the reactor hall. Also 25% of the halogens are released from the fuel

bundle (with the remainder assumed to plate-out on the relatively cool clad). As regarding the halogens that escape, 10% are assumed to form organic compounds which escape in the pool water. Only 1% of the balance is undissolved in the pool water and appears in the reactor hall air. The net halogen release to the reactor room and potentially outside is 2.725%. All other fission products remain in the pool or are otherwise unable to escape from the reactor room because of plate-out on cool surfaces.

3. CALCULATION AND RESULTS

Based on accident scenario and safety design of the nuclear reactor, can be calculated the amount of the core inventory released to the reactor pool, the fraction escaped from the pool into the reactor room, and the amount of radioactivity released into atmosphere directly from the reactor room or trough the stack. If we have a stack release, the efflux velocity of the gaseous releases can be expressed by:

$$w_0 = 1.274 \cdot \frac{R_{efflux}}{D^2}$$

where R_{efflux} is the release rate (exhaust rate from stack, from the design of the ventilation system) and D is the internal stack diameter. This efflux velocity will be used later to calculate the buoyant plume rise and the effective release height. The most commonly used atmospheric concentration calculation method is the Gaussian plume equation.

Taking into account our intention to find simplifying assumption in order to create nomograms for fast calculations, we will consider a nondepositing plume, and instead of time integrated concentration we will use the normalized surface concentration ($\chi u/Q$) which is a particular solution of the Gaussian diffusion equation.

$$(\chi u / Q) = \frac{\chi(x,0,z) \cdot u}{Q} = \frac{1}{\pi \sigma_y \sigma_z} \exp\left(-\frac{H^2}{2\sigma_z^2} \right) \quad (m^{-2})$$

The graphical solutions of the above equation are presented in Figure1 for B atmospheric stability class and several release height.
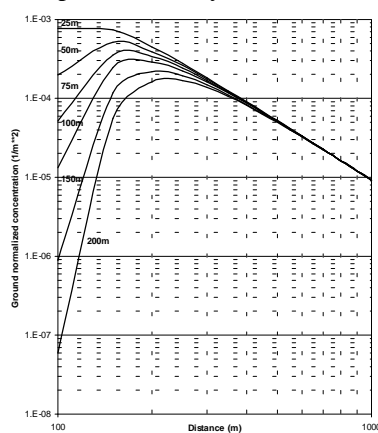
Stability class	Dist. (m)	Release height H (m)		
		25	50	75
A	150	0.994992	0.993518	0.99
	250	0.996973	0.996594	0.996
	500	0.998209	0.998123	0.998
	750	0.998638	0.9986	0.99
	1000	0.998871	0.998848	0.998
B	150	0.998328	0.997835	0.997
	250	0.99899	0.998863	0.998
	500	0.999403	0.999374	0.999
	750	0.999546	0.999533	0.99
	1000	0.999623	0.999616	0.999

Figure 1 - Ground normalized concentration for several release heights for B stability class	*Table 1* Plume dry depletion factor (DF) for **Iodine** (deposition velocity=0.07×10^{-2}, surface=soil)

After evaluation of the normalized surface concentration (χu/Q), the next step is to determine the activity Q released into atmosphere. This value must be corrected for disintegration, buildup and deposition. In this case we can write:

$$Q = Q_0 \cdot e^{-\lambda t} \cdot (DF)$$

where Q_0 is the total initial released activity, λ is the decay constant, t is the time elapsed from the release of contaminants and (DF) is the depletion factor accounting for the ground deposition of contaminants.

The correction for deposition must be calculated separately for dry and wet deposition. For dry and wet deposition:

$$(DF)_d = \exp\left[-\left(\frac{2}{\pi}\right)^{1/2} \frac{v_{dL}}{u} \int_0^x \frac{dx'}{\sigma_z(x') \cdot \exp\left(\frac{H^2}{2\sigma_z^2(x')}\right)} \right] \qquad (DF)_w = \exp(-\Lambda_L \cdot t_1)$$

where $\mathbf{v_{dL}}$ = average deposition velocity during the plume passage, t_1=time for the cloud scavenging effect (*s*) and $\mathbf{\Lambda_L}$ =scavenging coefficient (s^{-1})

Because here is intended only a simplified calculation, the contribution from any daughter from other isotopes will be not considered, and if we perform calculation for long lived radionuclide, we will not consider the exponential factor in equation above. Once we have evaluated the normalized ground concentration and total released activity corrected for disintegration and buildup, we can evaluate the doses for each organ of interest and for each pathway.

Any person immersed in a radioactive cloud will receive a certain cloud shine external dose. The dose to organ „o" received by an individual immersed in the plume can be calculated by:

$$D_{cld,o} = F_{Wext} \cdot \sum_k \chi_k \cdot (DCF)_{cld,o,k}$$

where: $D_{cld,o}$= the γ dose on organ „o", due to immersion in the cloud (*Sv*), F_{Wext} is the shielding factor acounting for time spent inside the buildings, χ_k is the time integrated concentration for „k" isotope ($Bq \cdot s \cdot m^{-3}$) and $(DCF)_{cld,o,k}$ is the dose conversion factor for „k" isotope and organ „o" for immersion in the cloud $[Sv/(Bq*s*m^{-3})]$

If we consider a single isotope, taking into account the definition of the normalized ground concentration, the corrected released activity and a shilding factor equal to 1, we can write:

$D_{cld,o}$=(χu/Q)·(Q/u)·(DF)·exp(-λt)·(DCF)$_{cld,o,k}$= (χu/Q)·(Q/u)·(DCF)$_{cld,o,k}$
$D_{Inh,o}$ =B (χu/Q)·(Q/u)·(DF)·exp(-λt)·(DCF)$_{Inh,o,k}$

The core has operated discontinuously for a total of 1780 MWd. Based on power operation history, and based on the composition of TRIGA fuel, we have constructed an input for ORIGEN computer code to evaluate the core inventory.

The result of the ORIGEN for core inventory of I^{135} is

Q_{core}= 2.7787 E+16 Bq (=7.51E+05 Ci)

the total released activity Q, corrected for disintegration, buildup and deposition is:

$$Q = Q_0 \cdot e^{-\lambda t} \cdot (DF) = 2.1025 \text{ E+13 Bq}$$

the ground normalized concentration for D stability class and a release height of 60m, at a downwind distance x=250 m from the stack, is

$(\chi u/Q) = 5.0$ E-04 (m^{-2})

Finally we can calculate the dose to a person at 250m from the stack, after 1 hour of immersion into the cloud:

$D_{cld,eff} = (\chi u/Q) \cdot (Q/u) \cdot (DCF)_{cld,eff,I}^{135} = 0.166$ mSv

The dose resulting from inhalation of radioactive materials is

$D_{Inh,eff} = B$ $(\chi u/Q) \cdot (Q/u) \cdot (DCF)_{Inh,eff,I}^{135} = 0.192$ mSv

If we consider as pathways for early exposure only the cloud immersion and inhalation, the total dose received by a person situated downwind at 250m from the stack, after 1 hour exposure is:

$\mathbf{D_{eff}} = \mathbf{D_{cld,eff}} + \mathbf{D_{Inh,eff}} = \mathbf{0.358}$ mSv

4. CONCLUSIONS

This dose calculated using a hand method will differ from those calculated by computer codes. A hand calculation of the dose received by a person in a certain location, following a radioactive release in the air, is a fast calculation method bat the result obtained is only an estimated value of the dose. The plume travel can be described in several manners depending on the degree of complexity of the model used. The plume rise and plume travel (including reflexions on the mixing layer) are complex phenomena and was not presented here. We have also to take into account that the sigma dispersion parameters and hence the dilution factor and time integrated concentration, strongly depend on the site location. Also some shielding factors were not considered (or were considered equal to unit). The only purpose of this calculation is to provide an example of how basic dose calculations can be performed manually.

REFERENCES

1.   Protection of the public in the event of major radiation accidents: principles for Planning, ICRP Publication 40, Ann. ICRP, 14 , No. 2, 1984.

2.   Planning for Off-Site Response to Radiation Accidents in Nuclear Facilities, Safety Series No 55, IAEA, Viena, 1981.

3. TECHNIQUES AND DECISION MAKING IN THE ASSESSMENT OF OFF-SITE CONSEQUENCES OF AN ACCIDENT IN A NUCLEAR FACILITY, IAEA SAFETY SERIES NO 86, IAEA VIENA, 1987.

4. COSYMA, A NEW PROGRAMM PACKAGE FOR ACCIDENT CONSEQUENCE ASSESSMENT, COMMISSION OF THE EUROPEAN COMMUNITIES, EUR 13028, BRUSSELS, 1991