Estimation of accidental environmental release based on containment measurements

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Abstract

In the past few years a new method for estimation of environmental release in a containment overpressure type accident at a nuclear power plant was elaborated in the Hungarian Academy of Sciences Centre for Energy Research. In case of large break loss of coolant accident a significant amount of radionuclides are released within a short period of time into the containment from the primary circuit.

The calculation method is based on measurements of the overpressure in the containment and the dose rate of radionuclides released to the containment. The initial rates of different nuclide activities in the containment can be estimated considering the gap activity and activity that may be released from the fuel matrix itself due to clad failure. Radionuclides released from the primary circuit can get partly to gas phase or liquid phase. This distribution rate is nuclide dependent, while the time dependence of the initial rate of different nuclide activities can be calculated by taking into account the radioactive decay, wash-out by the sprinkler system and deposition to the containment's walls. For the dose rate calculations the dose conversion factors for the position of the high range dose rate meter in the containment have to be known, for both gas and liquid phase.

In this way, based on the calculated dose rates for 100% fuel failure and on the measured dose rates, the actual activity concentration of each radionuclide in the containment's atmosphere can be estimated. The leakage emission from the containment is a function of overpressure that is also measured, thus the actual activity release rates from the containment can be estimated for each radionuclide.

An on-line, real time calculation code that performs calculations on the above method has been developed and put into operation at the Paks nuclear power plant.

Key words: LOCA, Nuclear Power Plant, Environmental Release

Introduction

In case of an accident in a nuclear power plant (NPP) the assessment of the consequence of the accident for the population is of high importance, so that the countermeasures, if necessary, can be taken. The first step of this assessment is the estimation of the activity release to the environment. The safety analyses of the NPP's for the different types of accidents are based on conservative assessments concerning both the activity release and the environmental dispersion situations.

In the case of a real accidental situation the estimation of the activity released to the environment should be based on actual measurements either by stack measuring devices and/or by measuring other characteristics of the reactor. Our paper describes a new method for determination of environmental release in the case of a containment overpressure type accident for pressurized water reactors. The method and a real-time calculation code for the Paks NPP have been elaborated in the Hungarian Academy of Sciences Centre for Energy Research in the past few years.

Activity Transfer From Failured Fuel To The Environment In Case Of Loca

One of the most important design basis accidents at a pressurized water reactor is the loss of coolant accident (LOCA), when a break occurs in the primary cooling circuit. In these cases a fraction of the core may fail and fission products accumulated in the gap between the fuel and the cladding can get to the primary circuit from the damaged fuels. Also activity may be released from the fuel matrix itself due to clad failure. The cooling water gets out to the containment through the break and a considerable part of it is immediately vaporized especially in cases of large break LOCA (LBLOCA). This transient results in a high-rate increase of the pressure in the containment. The fission products get partly into the containment air and partly to the liquid phase of the containment. A fraction of the radionuclides in the air phase of the containment can get into the neighbouring rooms by leakage due to increased pressure of the containment and eventually can get through room chains to the environment via the ventilation systems. The main stages of activity transfer are illustrated in Figure 1.

The method elaborated for assessment of environmental release is based on the measurements of two characteristic parameters, the dose rate from the fission products inside the containment and the overpressure.

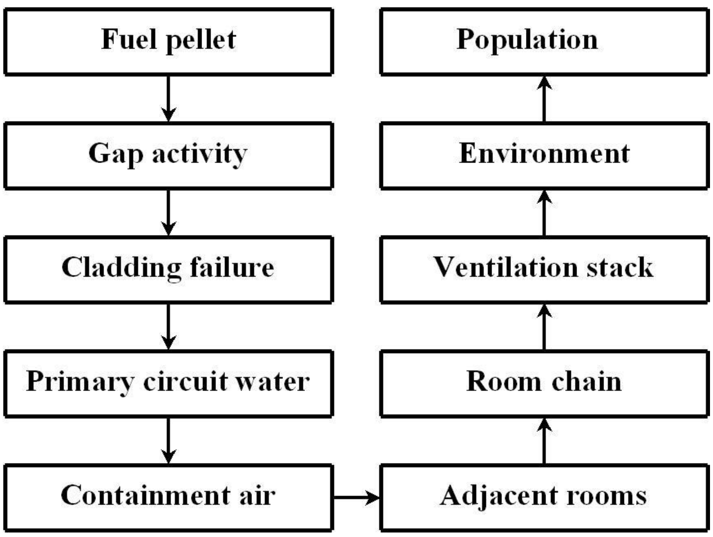

Figure 1. Logical scheme of activity transfer from failed fuel rods to the environment in case of overpressure type accident

Activity Released To The Containment

In the case of LOCA the rate of the fuel rods damaged in the accident is generally not known. In the present calculation method as a first step 100% failure is expected. The activity released to the primary circuit both from the gap and from the fuel matrix itself can be expressed as a rate of the core inventory in percentages. The total release is the sum of these two components. The details and parameters for these calculations are presented in the European Commission's recommendations for estimating the containment's source term in cases of large break LOCA [1]. Table 1 displays the core inventory at the Paks NPP, the percentages of the gap and the fuel release and the total activity released to the primary circuit for several radionuclides. Our calculation code applies the best estimate values instead of the conservative ones. In this case the gap release is restricted to maximum 1% and the fuel release to 6.5% of the core inventory, respectively. The maximum values are marked by an asterisk in Table 1.

Table 1: The core inventory of several radionuclides at the Paks NPP, the release rates from the gap and the fuel and the total activity released to the primary circuit in case of 100% fuel cladding failure

Nuclide	Half-life [hour]	Core inventory [Bq]	Best estimated values			
			Gap release [%]	Fuel release. [%]	Total release [%]	Total release [Bq]
^{85}Kr	94000	2.82E+16	1.00*	6.50*	7.50	2.12E+15
^{133}Xe	125,88	3.48E+18	0.20	1.95	2.15	7.48E+16
^{131}I	192.96	1.64E+18	0.25	0.40	0.65	1.07E+16
^{134}Cs	18100	3.10E+17	1.00*	0.86	1.86	5.77E+15
^{137}Cs	263000	2.27E+17	1.00*	1.00*	2.00	4.54E+15

In the code elaborated for the Paks NPP a data base for 53 of the radiologically most significant fission products are available. In the calculation it is assumed, that following the break in the primary circuit activities released from the gap and the fuel itself get instantly to the primary cooling water and then to the containment.

Assessment of Activity Concentration in the Containment

Activity released to the containment and activities calculated as described above – with assumption of 100% fuel cladding failure – is applied for assessing the initial activity rates of different radionuclides. Fission products released from the primary circuit can get partly to air phase or liquid phase (sump) of the containment. The initial distribution rates of nuclides in the air and in the liquid phase are taken from the EUR Report [1]. The starting time of the dry phase and the wet phase releases is closed up in our calculation method. Accordingly, it is assumed that 100% of noble gases, 57% of iodine (significantly in volatile I_2 form) and 10% of the other nuclides as aerosols (including iodine aerosol form) get to the containment atmosphere. The rest (apart from noble gases) is supposed to be carried to the containment sump. It is assumed that activity is promptly uniformly distributed in the air phase and the liquid phase of the containment.

The initial rates of nuclide activities relative to each other may be changed both in the air and in the sump of the containment due to different processes. The most important of such processes are the radioactive decay, the wash out of activity from air to the sump by the sprinkler systems and the deposition or condensation to the walls and water surface. For the total removal coefficients (for wash-out and deposition) from the containment atmosphere a conservative approach (as proposed in [1]) have been used.

Figure 2 illustrates the main processes affecting the activity transfer from the primary circuit to leakage trough the wall of the containment.

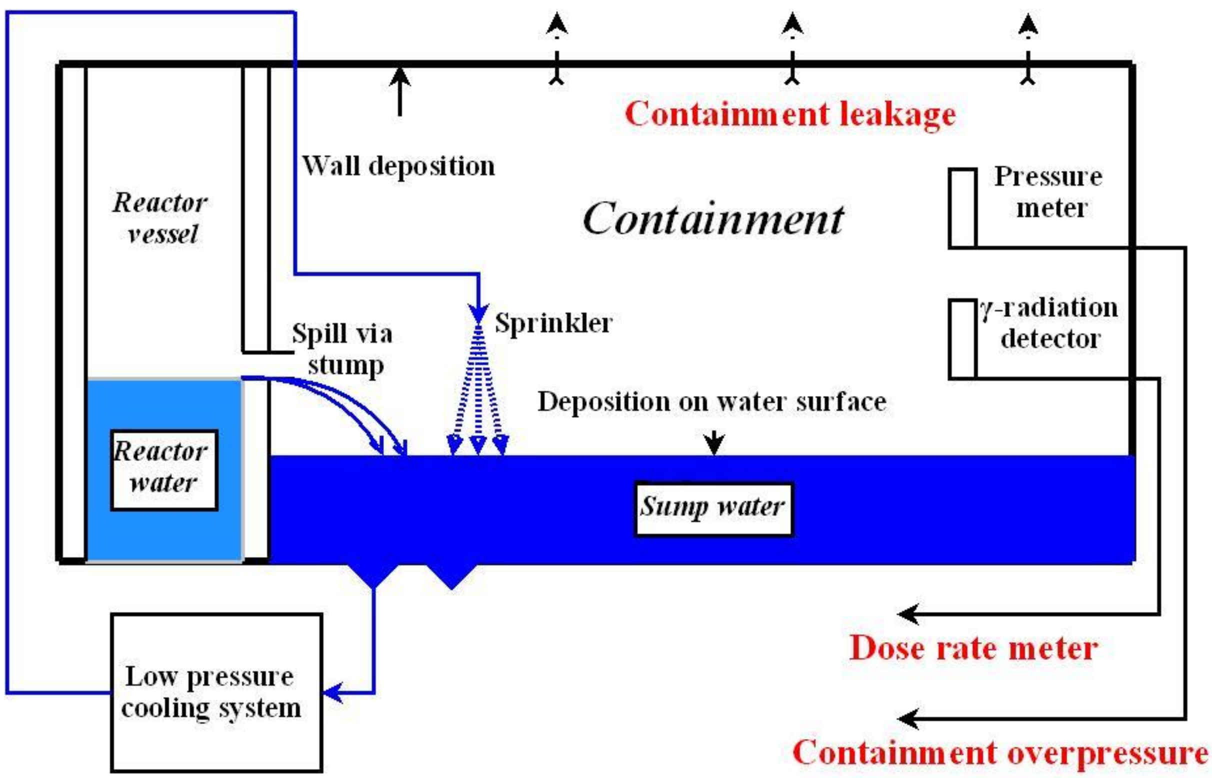

Figure 2 Activity transfer from the primary cooling circuit to leakage through the containment wall and the measuring devices applied by the calculation method

The water supply for the sprinkler system is ensured by the emergency core cooling system which also operates as a cooling system by circulating the water from the sump to the primary circuit.

The dose rate measured by the high range dose rate meter in the containment is the sum of the dose rates from the air and that from the sump. (The dose rate from radionuclides on the walls of the containment is neglected in the method and it is assumed that nuclides condensed on the walls get promptly to the sump.) Dose rates for the geometric position of the containment's dose rate meter were calculated by the Microshield program [2]. The dose rates for several nuclides in Table 2 were calculated in accordance with the initial assumption of the calculation method that the fuel rods have failed in 100% and fission products got promptly into the containment (at time of the break, t=0). In the dose rate calculations two different water levels of the sump were applied, 40 cm and 153 cm, respectively. At the Paks NPP the first sump water level refers to small break LOCA's, when the passive sprinkler is not activated. In the case of a LBLOCA a water backflow occurs from the trays of the localization tower due to rapidly increasing overpressure in the air above the trays' water. This backflow generally takes place within several minutes after the break. It is assumed that the water that flows back gets immediately to the sump.

The time dependence of the initial dose rates due to radioactive decay and removal processes from the air to the sump water are displayed in Figure 3.

Table 2 Total release to containment for several radionuclides in the case of 100% fuel failure and the initial dose rates from nuclides (and also the sum of all nuclides investigated) in the air and in the sump, for the cases of two different sump water levels at the time of the break, t=0

Nuclide	Total release to the containment [Bq]	Dose rate in containment [Gy/h]			
		from the air, for sump water level of		from the sump, for sump water level of	
		40 cm	153 cm	40 cm	153 cm
^{85}Kr	2.12E+15	5.37E-04	4.93E-04	-	-
^{133}Xe	7.48E+16	1.28E+00	1.18E+00	-	-
^{131}I	1.07E+16	3.15E-01	2.89E-01	1.11E-01	5.00E-02
^{134}Cs	5.77E+15	9.94E-02	9.14E-02	7.64E-01	3.44E-01
^{137}Cs	4.54E+15	2.96E-02	2.72E-02	1.86E-01	8.34E-02
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Sum for all nuclides		1.19E+01	1.09E+01	1.78E+01	8.02E+00

The leakage through the containment's wall due to overpressure concerns only radionuclides in the air phase of the containment. Nevertheless, the sum of the dose rate components both from the air and from the sump water are measured by the high range dose rate meter in the containment. A correction factor has been introduced for the measured dose rate to estimate the dose rate fraction from radionuclides in the air. The correction factor can be calculated according to Table 2 and Figure 3, namely, it is the quotient of the dose rates from nuclides in the air and nuclides both in the air and in the sump water. Then the air component of the dose rate can be estimated by multiplying the measured dose rate with the correction factor. The time dependence of this correction factor is represented in Figure 4 for the two sump water level investigated.

The actual activity concentration of radionuclides in the air of the containment can be estimated by normalizing the dose rate calculated with the assumption of 100% fuel failure to the dose rate measured in fact.

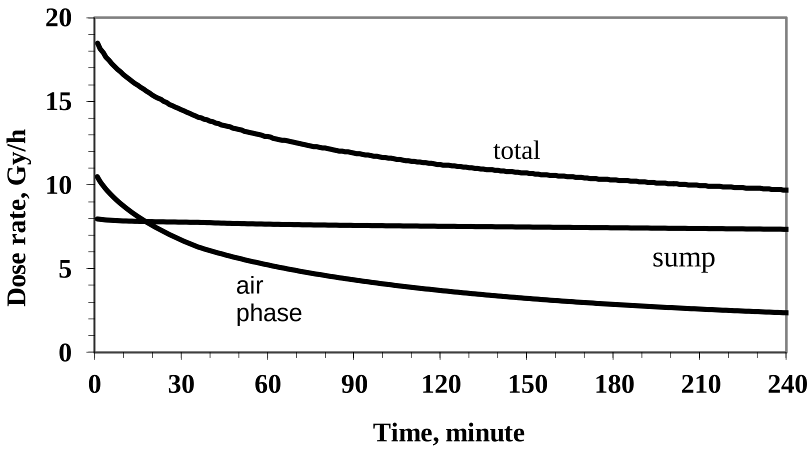

Figure 3 Time dependence of dose rates from nuclides in the containment's air and in the sump water as well as the total dose rate for the case of 153 cm sump water level

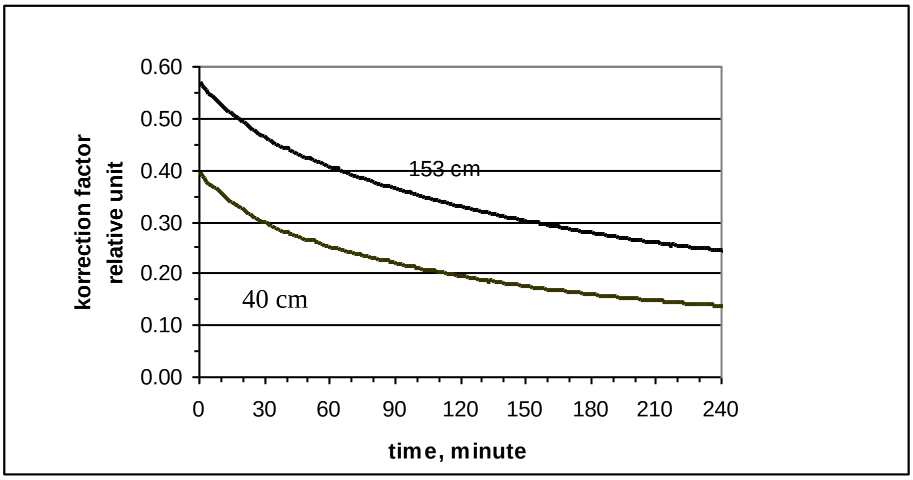

Figure 4 Time dependence of the correction factor for estimating dose rate fraction from nuclides in the air for two different sump water levels

Activity Transfer From The Containment To The Environment

The activity released from the containment is proportional to the activity concentration in the air phase and the leakage volume rate of air from the containment. The latter is periodically measured at the Paks NPP generally when the reactor is stopped for annual maintenance. In Figure 5 the leakage volume rate is presented as a function of the overpressure in the containment. The measured values were extrapolated to the maximum permissible overpressure and normalized for the maximum permissible leakage volume rate, these are 1.5 bar and 14.7% of the containment volume per day, respectively.

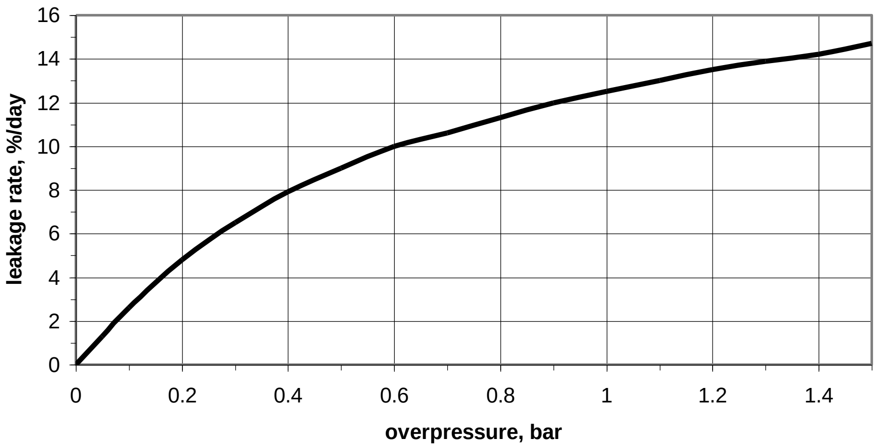

Figure 5 The leakage volume rate per day from the containment normalized for the maximum permissible value (i.e. 14.7% at 1.5 bar overpressure) as a function of overpressure in the containment

Thus, by using the yearly measured actual leakage volume rate of the containment, as well as the overpressure and the dose rate measured with a frequency of several seconds in the containment the activity release rate from the containment can be assessed.

Besides measuring the containment's total leakage rate also the distribution of the leakage between the neighbouring rooms and the filter efficiencies in the ventilation systems are measured in the course of the annual maintenance. The different possible routes of activity transfer from the containment wall to the environment are shown in Figure 6. The ventilation system of room or room chains concerned are equipped with

1) aerosol and iodine filters (for the elementary and organic forms of the latter),
2) aerosol filters only.

The activity can be released to the environment also without filtering:

3a) the ventilation system concerned does not contain any type of filter, 3b) the activity is not released via the ventilation stack of the NPP but through the rooms of the secondary circuit.

It is illustrated in Figure 6 that activity released to the environment via the ventilation stack may be measured by the PING (Particle, Iodine, Noble Gas) measuring devices, but the activity is released through the rooms of the secondary circuit is unmeasured. Nevertheless, simulation calculations have attracted attention to cases of large break LOCA, when the PING measuring devices, especially the iodine unit may be saturated. Our new calculation method enables to assess the activity released to the environment even if direct measurements cannot be performed.

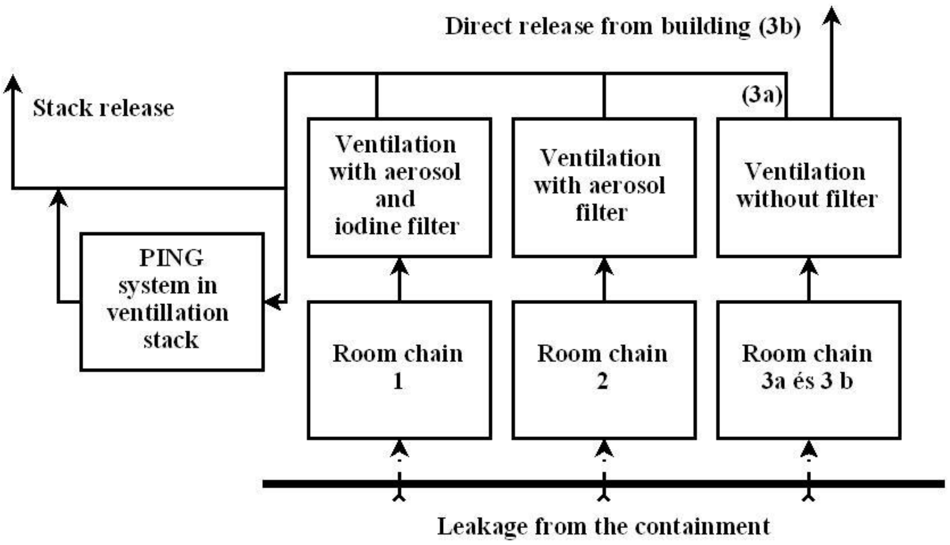

Figure 6: Possible routes of activity release to the environment from the outside wall of the containment

Results

Based on the above described calculation method a real-time on-line computer code has been put into operation at the Paks nuclear power plant. A logical scheme for calculating the activity released from the containment and the activity transfer from containment to the environment is shown in Figure 7.

The on-line computer code is equipped with a user-friendly comfortable and transparent interface. Via this interface the user can provide the code with the yearly measured actual total leakage volume rate of the containment and the distribution of the leakage between the neighbouring rooms. All the other input data are selected and forwarded to the computer in one minute data blocks from the main information and data storage system of the NPP.

When no LOCA event occurs, the program works as a pre-processor. It checks the time stamps of the consecutive data blocks, identifies data by their alphanumeric identification code, checks the validity of data by their status stamps, moreover, it performs preliminary data evaluation for LOCA calculations and stores input the database for the last 48 hours. All these input data can be displayed graphically by the visualization module.

Activity transfer calculations are started automatically if the overpressure exceeds 3 mbar in any containment of the four reactor blocks. Parallel LOCA calculations are allowed for the reactor units. LOCA calculations are executed for 48 hours after the overpressure is ceased. In the course of the event results from the beginning to the actual time can be displayed both graphically and numerically. The parameters of LOCA calculations and the results are stored for possible further analysis. Output data of the LOCA calculations listed below refer for ten minute intervals:

- differential and integrated release of individual radionuclides or nuclide groups via the ventilation stack and/or the buildings of the secondary circuit,
- dose rates (external gamma, committed effective inhalation and thyroid dose rates for the last ten minutes) in rooms concerned by the route of activity transfer,
- comparison of measured data and calculated values for rooms where dose rates are measured and for the stack's release measuring devices.

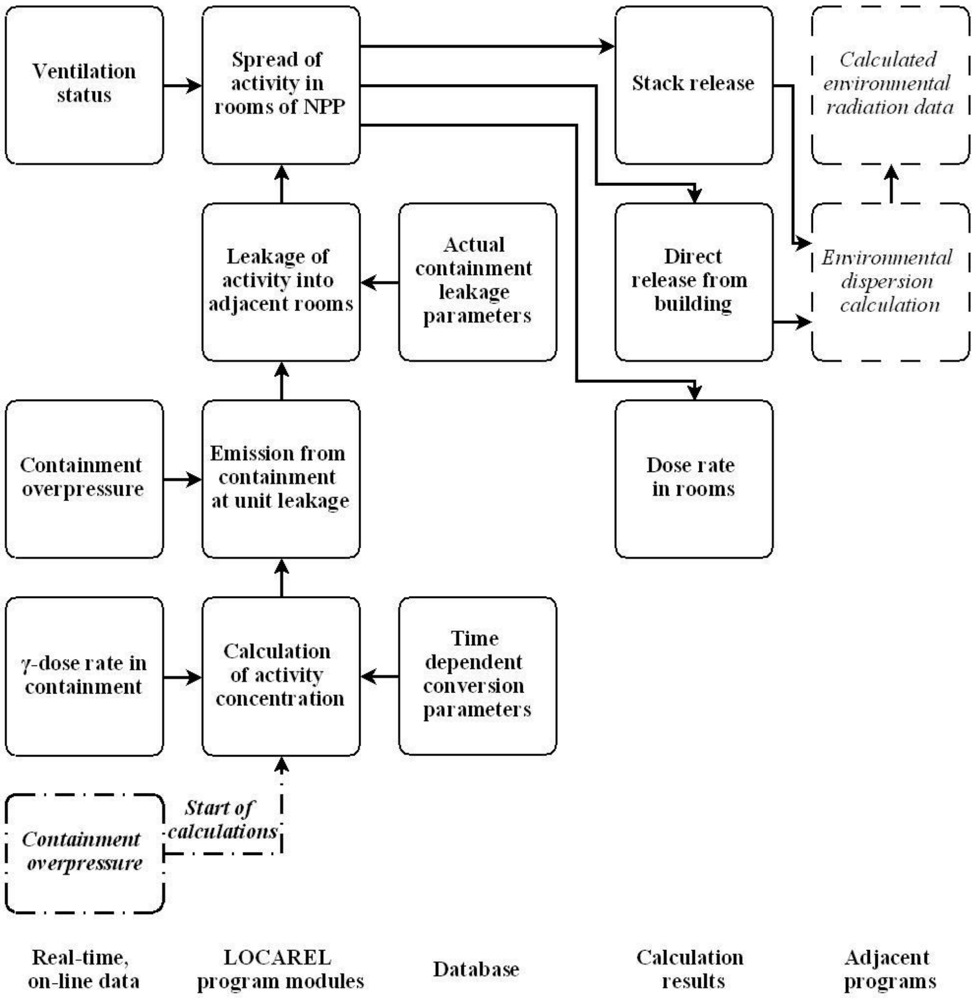

Figure 7 Logical scheme for calculation of the activity released from the containment and of the activity transfer from containment to the environment

References

[1] Determination of the in-contaiment source term for a Large-Break Loss of Coolant Accident; Europena Commission; EUR-19841 EN; 2001

[2] MicroShield Users Manual version 6.20; Grove Software Inc; 2005