Estimation of accidental environmental release based on containment measurements

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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.

Description

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.

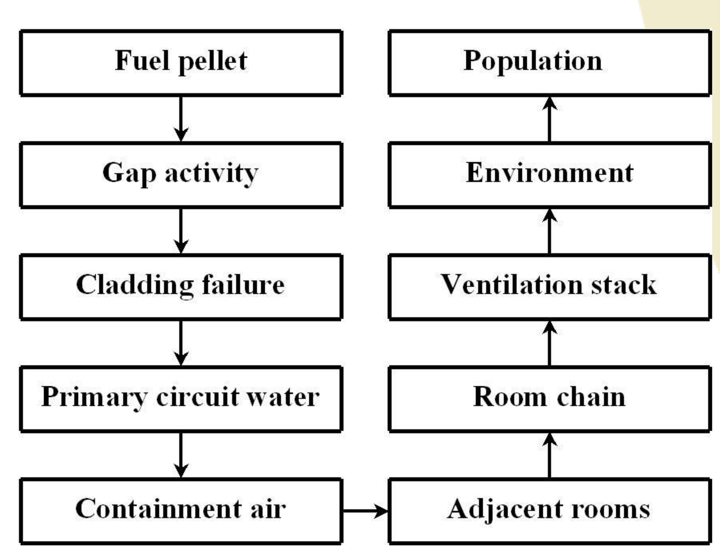

Fig. 1: Logical scheme of activity transfer from failed fuel rods to the environment in case of overpressure type accident

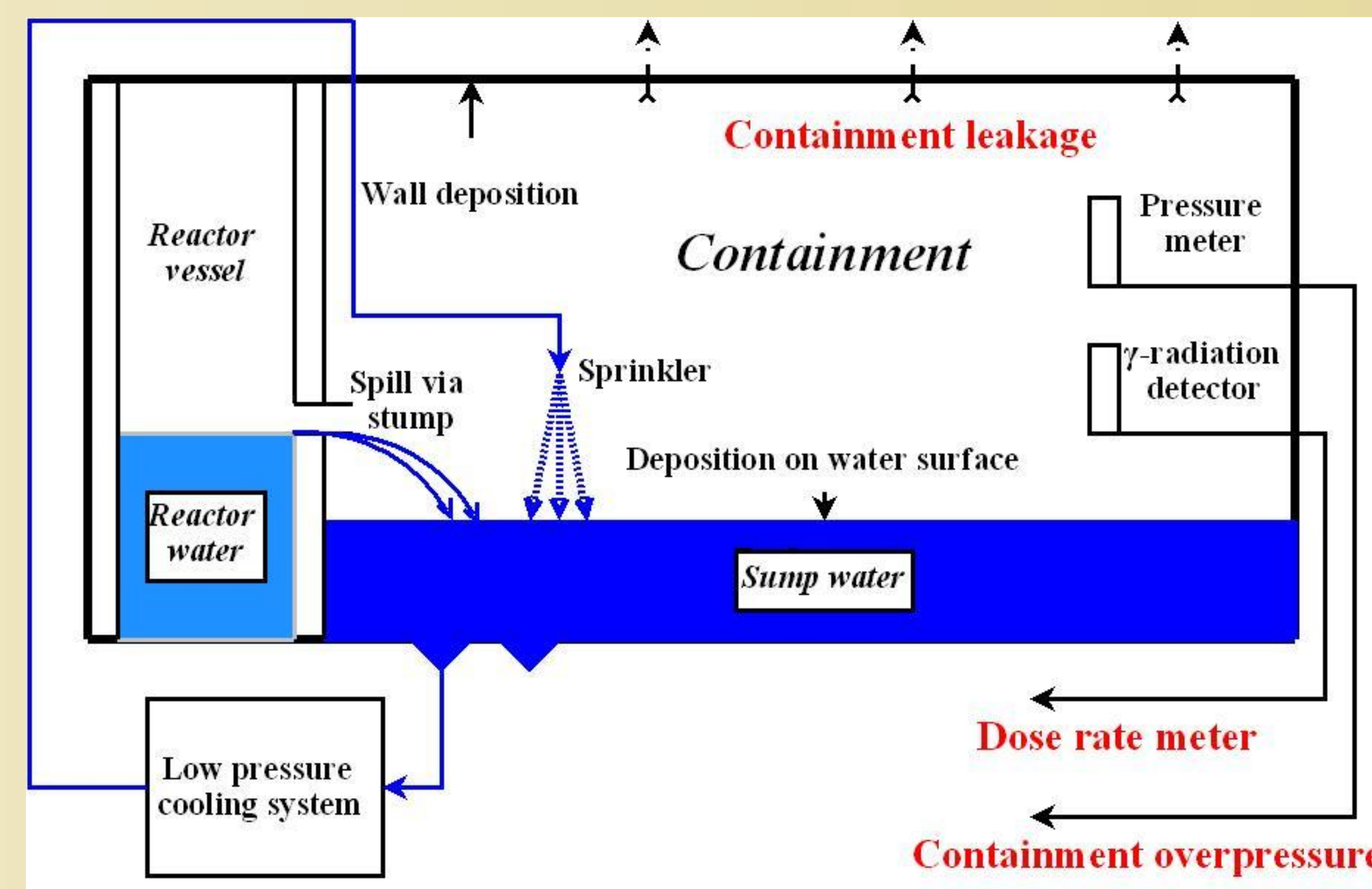

Fig: 2: Activity transfer from the primary cooling circuit to leakage through the containment wall and the measuring devices applied by the calculation method

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 (Fig. 1, 2 and 3) is a function of overpressure that is also measured, thus the actual activity release rates from the containment can be estimated for each radionuclide.

Result

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 Fig. 4.

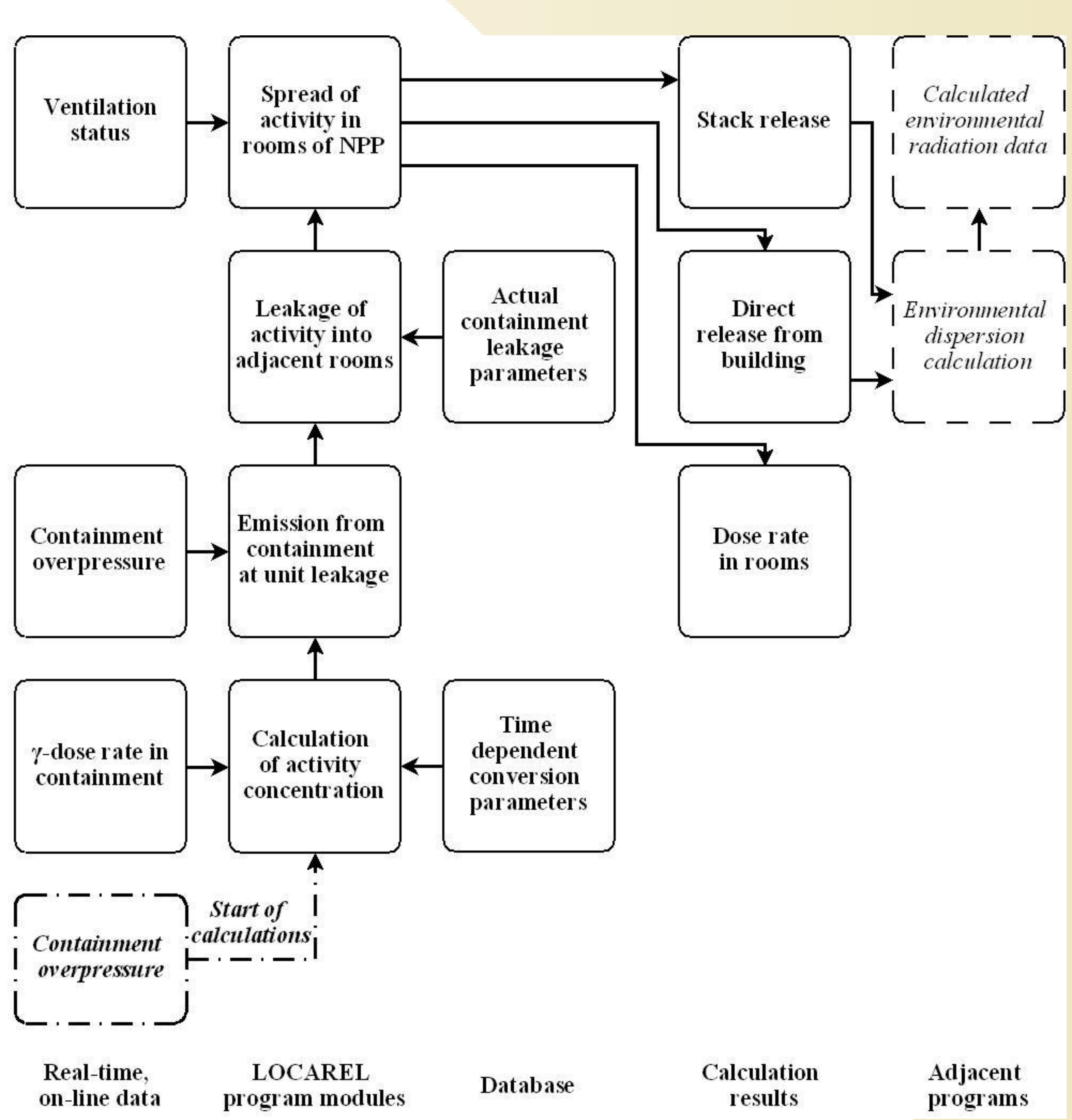

Fig. 4: Logical scheme for calculation of the activity released from the containment and of the activity transfer from containment to the environment

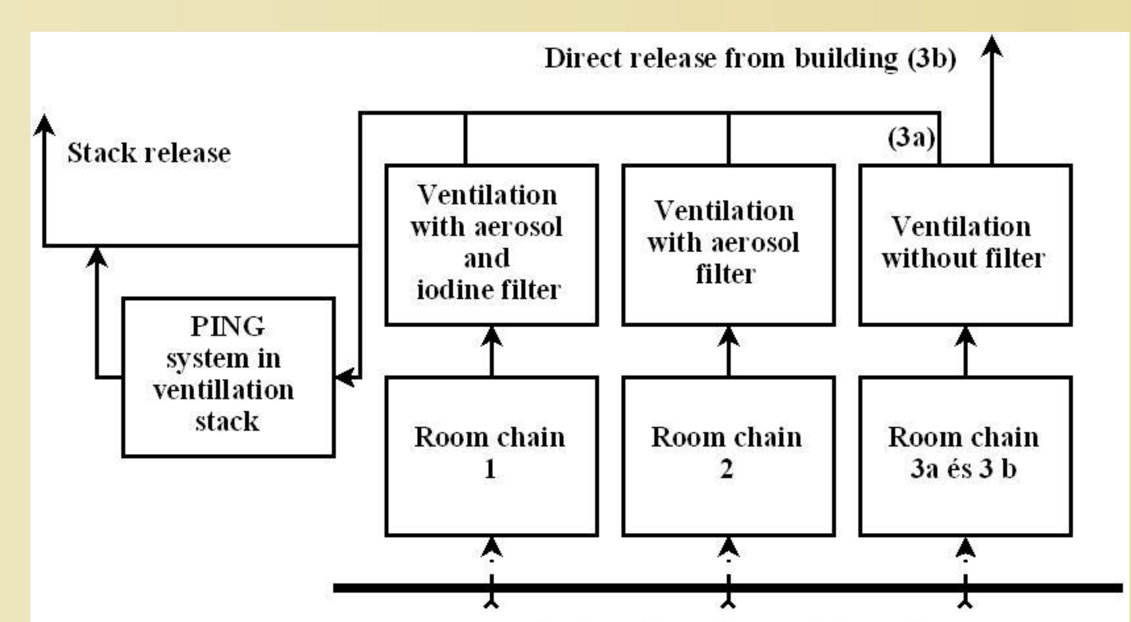

Fig. 3: Possible routes of activity release to the environment from the outside wall of the containment

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. LOCA calculations are executed for 48 hours after the overpressure is ceased. 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 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.

References

[1] Determination of the in-containment 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