CA9800116

SOURCE TERM ASPECTS ASSOCIATED WITH FUTURE PWR CONTAINMENT SYSTEMS

B. Kuczera, G. Keßler, J. Ehrhardt, W. Scholtyssek

ABSTRACT

The overall objective of reactor safety is to protect the population against dangerous releases of radioactive materials from nuclear power plants. In context with a reinforcement of the defense-in-depth strategy the common safety requirements on future nuclear power plants converge in the objective that these plants should be so safe that even in case of a severe accident there will be no need of off-site emergency actions such as an evacuation or resettlement of the population from the vicinity of a nuclear power plant. It is shown by the example of a future 1400 MW_e pressurized water reactor (PWR) plant that this goal can be attained in principle by providing a double containment with the annulus vented via an appropriate emergency standby filter. Within the framework of severe accident consequence mitigation a set of parameters for accident conditions and emergency filter efficiencies is elaborated under which the German lower levels of intervention for evacuation are not attained.

Nuclear Research Center Karlsruhe (KfK)
Postfach 3640, D-76021 Karlsruhe

SOURCE TERM ASPECTS ASSOCIATED WITH FUTURE PWR CONTAINMENT SYSTEMS

B. Kuczera, G. Keßler, J. Ehrhardt, W. Scholtyssek

1. INTRODUCTION

The overall objective of reactor safety is to protect the public and the environment against dangerous releases of radioactive materials from nuclear power plants. In a modern 1400 MW_e pressurized water reactor (PWR) the radioactive inventory of its equilibrium core amounts to approximately 10^{21} Bq. In order to assure a safe confinement of this inventory a staggered multi-barrier system has been established whose confinement function is designed such that in case of failure of one barrier the radioactive materials are confined by the next physical barrier. Figure 1 illustrates once more this concept which includes the fuel matrix, the fuel cladding and the reactor coolant system. The ultimate barrier of this preventive system is the reactor containment, here represented by a spherical steel shell. The shell is protected from external impacts by a reinforced concrete structure which serves at the same time as a secondary containment. The annulus between the concrete wall and the inner steel containment is permanently exhausted via a particulate in air filter; in case of an accident the exhaust is via a special combination of particulate in air and iodine filters which will largely exclude that radioactive materials leaking from the steel containment are directly released into the environment. An alternative containment system is represented by a double concrete cylinder design with a similar annulus between both structures.

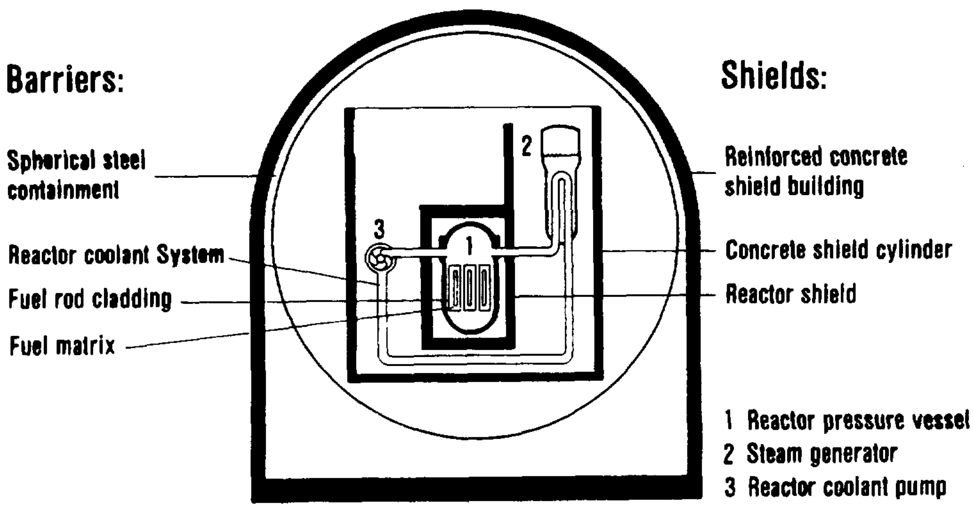

Fig. 1: The staggered multiple-barriers concept against accidental release of radioactive materials illustrated by the example of a PWR plant.

In the recent past numerous improvements have been proposed for future PWR systems, the common basis of which is the requirement to consider severe accident load conditions already at the plant design stage [1]. A general rationale for these innovative trends is outlined elsewhere [2], where the new safety objective reads like this: In future PWR's, even a core meltdown accident and its consequences should not require off-site emergency actions such as an evacuation or a resettlement of the population from regions in the vicinity of the damaged plant. This leads to a new safety quality the demonstration of which has to be performed in a deterministic way, i.e. on the basis of best engineering judgement; complementarily, probabilistic assessments should demonstrate the balanced approach of the overall safety concept [1, 3].

In this context, the radiological source term - or more precisely - the confinement of the radiological source term is of decisive importance and related aspects have therefore been placed in the foreground of this contribution. The radiological source term represents the portion of radioactive core materials which is assumed to be accidentally released from a nuclear power plant into the environment and which has to be minimized in order not to exceed the threshold dose values for off-site emergency action to be taken. In this respect parametric investigations were performed which indicate plant design characteristics to attain this goal.

2. REFERENCE PLANT CONDITIONS AND SEVERE ACCIDENT SCENARIO

For the subsequent investigations an advanced 1400 MW_e PWR plant with a double containment has been chosen as reference case, the global reactor data of which are specified as following [4, 5]:

Thermal reactor power	= 4250 MW (equilibrium UO_2 core conditions)
Initial fuel enrichment	= 4.5 Wt - % U235
Fuel cycle length	= 292 full-power-days
Average discharge burnup after six fuel cycles	= 60,000 MWd/tHM
Activity inventory after core shutdown	= $9 \cdot 10^{20}$ Bq
Activity inventory after 20 days	= $6{,}5 \cdot 10^{19}$ Bq

Within this frame work certain assumptions have to be introduced concerning the leakage of the primary containment under accidental pressurization. For a steel containment a leak rate of 0.25 Vol. %/d is usually anticipated and for a prestressed concrete containment the corresponding value is 1 Vol. %/d. For reasons of conservatisms the parametric study on radiological consequences has been extended up to a maximum leak rate of 1.5 Vol. %/d. In this context it is further assumed that containment bypass sequences (often called "V-sequences") can be practically excluded by appropriate design measures [2], i.e. radioactive leakages from the primary coolant circuit through connected auxiliary systems located outside of the containment are not considered.

For the following studies a low-pressure core meltdown scenario (LP path) is supposed. Accordingly, the core starts melting about one hour after the onset of a loss-of-coolant accident, and after another hour it causes failure of the reactor pressure vessel. Core meltdown is a complex multi-component process in physical/chemical terms. With ongoing core heating in the temperature range between 1500 °C and 2500 °C the core meltdown spreads to all core components. During this process the following portions of the core activity inventory are supposed to be released into the containment atmosphere during core meltdown :

noble gases, halogens, alkalis, silver	100%
antimonum, barium	50%
strontium	30%
tellurium, selenium	25%
zirconium	3%
ruthenium, lanthanum, cerium	2%
transuranium elements	0.3%

For the theoretical analysis it is conservatively assumed that during core meltdown a total mass of about 3.5 t of corium aerosols are released from the primary circuit which are homogeneously distributed in the containment atmosphere . The aerosol behavior in the containment is significantly influenced by physical separation mechanisms such as coagulation, sedimentation and condensation which cause the permanent reduction of the aerosol concentration in the accident atmosphere . This behavior is simulated by the aerosol model MAEROS which is included in the containment code system CONTAIN [6, 7]. For subsequent investigations the following input parameters have been used:

<u>Containment</u>

Free volume	71,200 m^3
Surface of the internal structures	50,000 m^2
Sump water volume	1,600 m^3
Parameter: Leak rate into the annulus	0.25/0.5/1.0/1.5 Vol. %/d

<u>Aerosol data</u>

Mass released from the primary system	3.46×10^3 kg (3% radioactive)
Median aerosol particle diameter	0.15×10^{-6} m
Geometric standard deviation	2.0
Density of the aerosol material	5000 kg/m^3

The core meltdown accident is simulated in the CONTAIN model by

(a) the corium aerosol and noble gas sources, as described earlier which are released into the containment; their releases are assumed to take place within the time interval of 30 to 100 minutes after onset of the accident;

(b) the direct transfer of the decay heat from the core melt into the sump water of the containment (i.e. core melt cooling by sump water in a "core catcher device" is assumed).

The development of pressure and temperature in the containment is determined mainly by the decay heat released from the core melt into the sump water and the heat removal from the containment to the environment (20°C). <u>Figure 2</u> shows plots of the time dependent decay heat power, pressure and temperature versus time. According to these conservative analyses a pressure rise to about 5.5 bar has to be anticipated to take place within the first 80 h, with a corresponding temperature of about 145°C to it.

The corresponding aerosol behavior is represented in <u>Figure 3</u>. The dashed line shows the airborne aerosol mass in the containment as a function of time. It is clearly evident that already during the early aerosol release phase (a linear model was used here) considerable amounts of the total mass of 3,5 tons are separated from the atmosphere through agglomeration and sedimentation. The maximum value of airborne aerosols is about 2.1 t at the end of accident initiation phase (t = 100 min) which corresponds to an aerosol concentration of about 30 g/m^3. During the following 10 hours the mass of airborne aerosols in the containment is reduced by about 3 - 4 orders of magnitude. This means for the aerosols leakage from the inner containment that this process has largely vanished at the end of the 10 - 12 hours period. This is evident also from the curves plotted in <u>Figures 3</u>

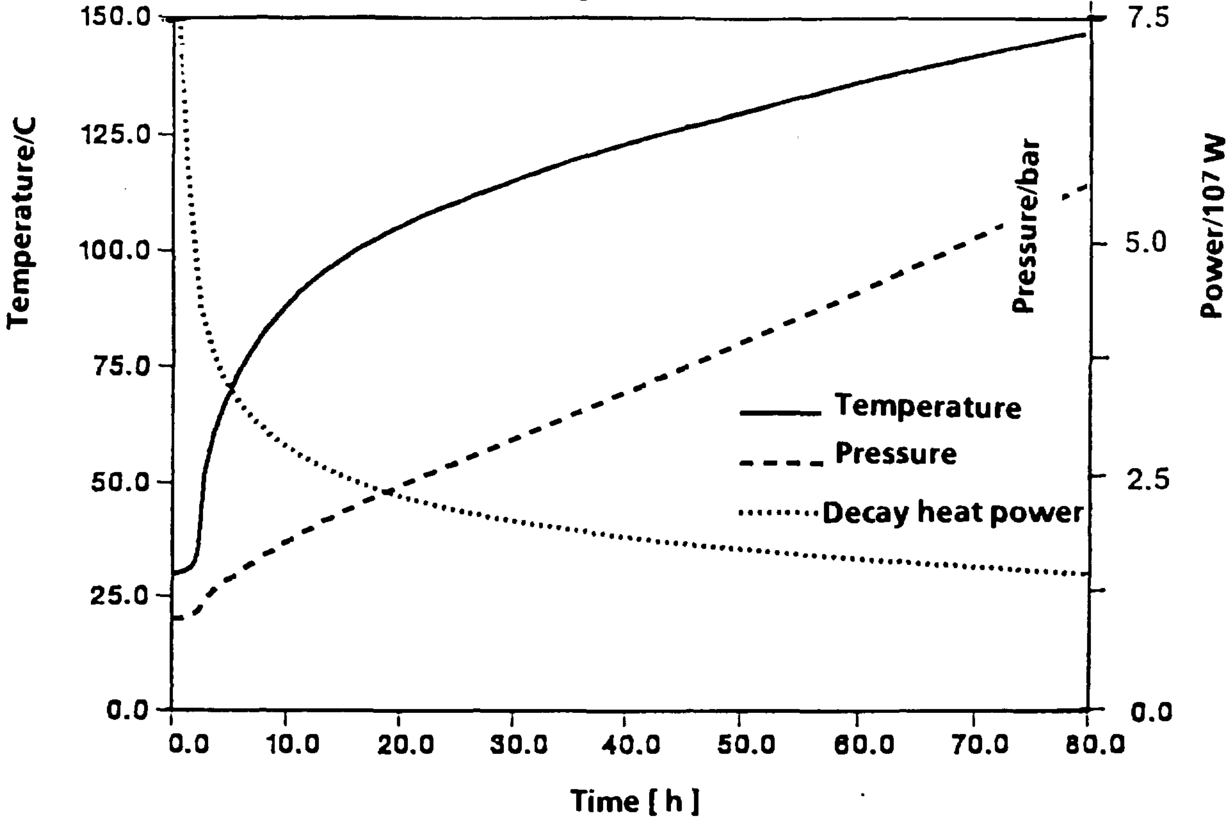

Fig. 2: Time dependent decay heat power released into the containment atmosphere and corresponding long-term pressure and temperature behavior.

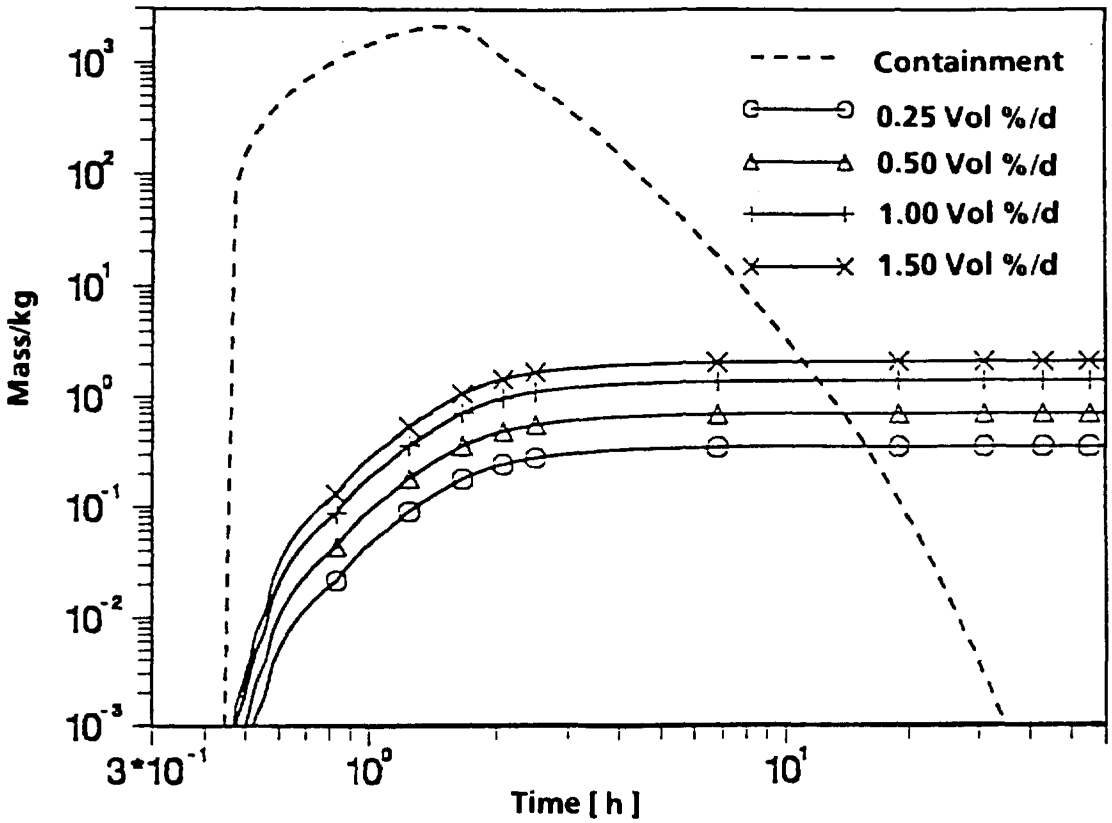

Fig. 3: Airborne aerosol masses in the containment atmosphere and aerosol leakages into the annulus (Parameter: the inner containment leak rate).

reflecting the loss of aerosols from the inner containment at different leakage rates. The model computations yield as tentative leakage values for this time span about 0.36 kg in total (corresponding to about 0.01% of the entire aerosol mass produced) at a low leak rate and a total of about 2.2 kg aerosols which have escaped from the primary containment into the annulus at 1.5 Vol%/d. Additionally it should be mentioned that in these computations the condensation of vapor onto the aerosol particles which promotes aerosol separation from the atmosphere has not been taken into account so that the results can be considered conservative with regard to the radiological leakages.

The leakage behavior of the radioactive noble gases Xe and Kr is quite different. The values calculated with CONTAIN represent portions of the core inventory. Also in this case the release into the containment takes place during the interval t = 30 - 100 minutes, as shown by the dashed line in Figure 4 This Figure makes evident that the leakage of noble gases (integral values) from the containment into the annulus increases almost linearly with the time, however with different slopes, depending on the leak rate. A more detailed analysis of the impact on the radiological source term will be given in the next Chapter.

But before that, another aspect is briefly addressed which relates to the complex iodine (I) behavior in the sump water. From the radiological point of view the behavior of iodine - especially of I-131 - is of particular importance in terms of its

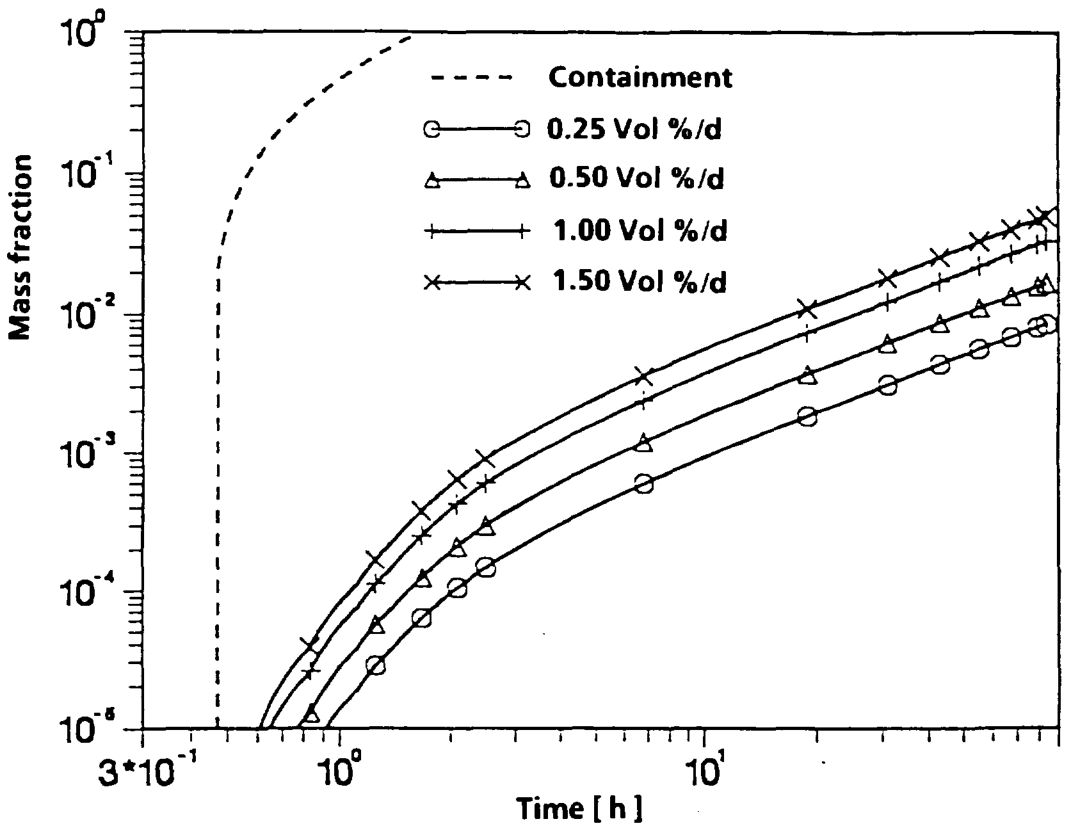

Fig. 4: Mass fractions of the radioactive noble gases (Xe, Kr) in the containment atmosphere and noble gas leakages into the environment (Parameter: the inner containment leak rate).

environmental burden. When released from the fuel into the containment, iodine preferably reacts with cesium (Cs) so that initially about 99 % of iodine is present as CsI and about 1 % as elemental iodine I_2.

The aerosols suspended in the atmosphere of the containment together with the fission products contained in them, among them CsI and AgI, are separated by sedimentation, diffusion and other physical processes.When present in the liquid phase, CsI immediately dissociates completely into Cs^+ and I^- ions while AgI remains undissolved. I_2 exists in liquid and gaseous phases, which are equally distributed. Against the background of the complex physico-chemial behavior of these fission products rather conservative assumptions were chosen for the off-site accident consequence assessments.

3. OFF-SITE ACCIDENT CONSEQUENCE ASSESSMENTS

3.1 Boundary Conditions

Probabilistic accident consequence assessments were performed on the basis of 144 representative weather sequences using the COSYMA program package [8, 9], version 92/1, which is European wide applied and accepted. The weather sequences were selected by stratified sampling from synoptic hourly recordings of KfK in the years 1982 and 1983. For the diffusion calculations with the MUSEMET Gaussian trajectory model implemented in COSYMA, the Karlsruhe-Jülich σ-parameters, attributed to roughness class 3, were used which are contained in the German regulations as well. The dose reference levels at which intitiation of the emergency actions "sheltering", "evacuation,"and "relocation" are recommended are given in Table 1. For the measure "restrictions in the distribution of food" , the maximum permissible levels of activity concentrations published by the CEC are valid [10] . The intervention dose levels for emergency actions used in all computations were the lower reference levels indicated in Table 1. The calculations are based on the activity inventory in the containment atmosphere, related to the time of the accident (end of chain reactions) as indicated in the previous Chapter. For the chemical forms of iodine the assumption is made that 5% occur in elemental form, i.e. as I_2 gas, and 95% as aerosols in the containment atmosphere. A leak rate of 1.5 Vol.%/d is assumed for the inner containment. Regarding the release due to leakages from the annulus via the stack into the environment, a distinction is made between filtered and unfiltered release of leaking substances.

As release height, the stack opening at 180 m is assumed. The wet and dry deposition parameters are identical with the COSYMA default values. Dry deposition velocity for aerosols is 10^{-3} m/s, for elemental and organically bound iodine, the values 10^{-2} m/s and $5 \cdot 10^{-4}$ m/s are used.

3.2 Assessment Results

Probabilistic accident consequence assessments in principle provide frequency distributions of the various types of consequences. The individual radiation doses of interest in this context are generally represented as a function of the distance, with not the distributions proper indicated, but rather the statistical variables derived from them, such as mean values or percentiles. The 95%-fractile of a dose

Measure	Dose [mSv]					
	Whole Body**) Lower reference level	Upper reference level	Thyroid Lower reference level	Upper reference level	Lung*) Lower reference level	Upper reference level
Sheltering	5	50	50	250	50	250
Distribution of iodine tablets	-	-	200	1000	-	-
Evacuation	100	500	300	1500	300	1500
Relocation	50	250	-	-	-	-

*) or each preferably exposed single organ except for the skin
**) actually: effective dose

Tab. 1: Dose reference levels for emergency actions in Germany

frequency distribution, for example, gives the dose value, which is not exceeded in 95% of all accident consequence situations.

3.2.1 Radiation Doses Resulting from Mere Noble Gas Release (Idealized)

Unlike the aerosols, the noble gases are not deposited in the containment. The activity concentration is reduced exclusively by radioactive decay.

To quantify the period during which the noble gases contribute substantially to the individual dose value outside the containment with an assumed leak rate of 1.5 Vol.%/d, hourly releases with a released fraction of 6.25×10^{-4}/h of the activity inventory were considered over an interval of 48 hours. This value results from the leak rate cited above and the assumption made that 100% of the noble gas inventory is present in the containment atmosphere from the very beginning.

The results show, that due to radioactive decay of the short-lived noble gas nuclides the individual dose has accumulated to roughly 90% after about 10 hours. During the first hour the main contribution to the radiation dose stems from Kr-88 (42%) and Xe-138 (37%).

	400 m	1000 m
95%-fractiles [mSv]	0.65	0.39
99%-fractiles [mSv]	2.1	0.81

Table 2: Percentiles of intervention doses (effective) at two distances

Even with an aerosol and iodine retention of 100%, the radiation doses from noble gas release could not be avoided. It can be seen from Table 2 that with 1.5 Vol.%/d leak rate due to the noble gases alone the safety margin between the lower reference value for sheltering (5 m Sv according to Table 1) and the dose values is relatively small, staying in houses as a protective measure cannot be excluded. This could only be avoided with high probability solely by providing an inner containment whose leak rate remains well below 1.5 Vol.%/d, even at high internal pressure.

3.2.2 Radiation Doses Resulting from the Combined Release of Aerosols, Iodine and Noble Gases

Due to the relatively short time of accumulation of the radiation doses resulting from noble gases of about ten hours and the almost completed release of aerosols and iodine at the end of 10 hours (Fig. 3), the source term was modelled for all radionuclides in the same way by six release phases of one hour duration. The dose was calculated in conformity with the regulations for determination of intervention doses. This means that the results can be directly compared with the intervention values for emergency actions according to Table 1. First analyses show that with the containment built as a single cylinder certain limit values indicated in Table 1 are exceeded. Therefore, besides the single cylinder containment also a double-cylinder containment is being studied in which the space between the two cylinders is exhausted via an aerosol and iodine filter.

The releases from the single-cylinder version are termed "unfiltered," the releases from the version with intermediate space and exhaust system are termed "filtered." Figure 5 shows the 95%- and 99%-fractiles of the effective dose equivalent (in short effective dose) as a function of the distance, with and without filtration of aerosols and iodine (filter efficiencies 99% for aerosols and 90% for I_2).

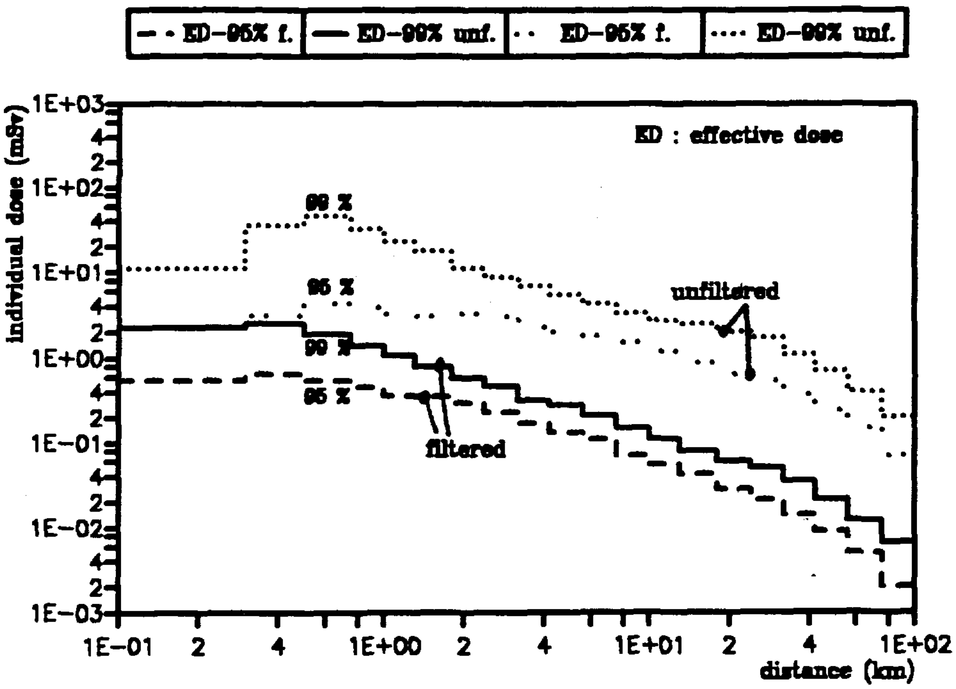

Fig. 5: 95%- and 99%-fractiles of the effective doses as a function of distance (intervention doses for early emergency actions).

In case of filtered releases the highest effective dose values occur within a distance up to about 1 km; they have been entered together with the values for the dose equivalent for the thyroid in Table 3:

	Unfiltered		Filtered	
	95%-fractiles	99%-fractiles	95%-fractiles	99%-fractiles
Effective dose [mSv]	4	46	0.64	2.5
Thyroid dose [mSv]	20	340	1.0	6.6

Table 3: Percentiles of intervention doses at about 1 km distance

The comparison with the lower reference values according to Table 1 makes clear that because of the small safety margin between the lower reference levels for intervention and the calculated intervention dose values in case of unfiltered release the necessity of emergency actions (sheltering, distribution iodine tablets, evacuation) cannot be excluded; in case of filtered release, on the other hand, the expected doses remain clearly below the lower reference values, and in none of the accident consequence situations evacuation areas are calculated.

Similar results were obtained regarding relocation measures. In case of unfiltered release dose values may occur in the range of the lower reference values for relocation. On the other hand, filtered releases cause in 99% of the accident consequence situations radiation doses lower by approximately two orders of magnitude than the lower reference level for relocation. No relocation areas were calculated for the accident consequence situations considered.

Concerning the need for agricultural countermeasures, the results show that in case of filtered release no food bans lasting more than two years are imposed in more than 99% of the cases; the 99%-fractiles of the areas affected are smaller than 4 km^2 for a two year ban with the maximum value for potatoes. The largest areas calculated are for milk and dairy products; the 99%-fractiles are about 720 km^2 whereas the duration of the measure is less than three months.

In case of unfiltered releases, however, food-bans for milk and milk products expand up to 9800 km^2 (99%-fractile); the largest areas are calculated for sheep meat with a 99%-fractile of about 10.000 km^2. In the second and fifth year food bans are calculated for areas up to 4600 km^2 (potatoes) and 500 km^2 (sheep meat), respectively (99% fractile). For sheep meat an area of about 40 km^2 is still affected by foodbans in the tenth year.

Given the uncertainties concerning the amounts of the various chemical forms, the pessimistic assumption has been made in supplementary dose assessments that 100% of the released iodine occurs as elemental iodine (I_2). The results show that there is an only small increase in the dose values, the statements made as regards the necessity of emergency actions continue to be valid.

There are discussions as to whether by dynamic events or turbulences major amounts of aerosols get resuspended in the containment atmosphere so that they cause releases over extended periods of time. One example in this context is considered to be the combustion of hydrogen and its detonation, respectively, or an ex-vessel steam explosion.

Consequently, it has been supposed in a pessimistic assessment that during a period of 48 h no aerosol deposition takes place in the containment and that noble gases as well as aerosols and iodine are released filtered via the stack. It has appeared that in the absence of aerosol deposition in the containment, even after 48 h release through leakage, the resulting radiation doses are well below the lower reference levels for evacuation in the "filtered case".

4. CONCLUDING REMARKS

A radiologically adequate insulation of future PWR plants against the environment with the objective of limiting accident consequences to the plant itself, even in case of extremely unlikely core meltdown accidents, so that from the technical point of view there will be no necessity of off-site emergency planning and evacuation of the population, respectively, calls for the following engineering measures:

- A double containment system whose inner shell maintains its integrity even under extreme accidental load conditions.
- The leak rate of the inner containment should not exceed the typical value of 1.5 Vol.%/d (even under high internal pressure loading).
- Leakages from the inner containment shall be accumulated in the annulus and discharged in a controlled manner via an annulus filter system and the stack. The desired efficiency of the filter is 99.9% for aerosols and 99% for elemental iodine (it must be examined whether passive stack draft will be able to achieve sufficient exhaust of the annulus).

As to the radiological risk potential, the investigations have shown that it clearly diminishes within the first 10 to 12 hours following the onset of an accident. These are the reasons:

(i) The radioactive decay of the noble gases, especially the short-lived radionuclides such as Xe-138 and Kr-88, which initially contribute by 37% and 42%, respectively, to the radiation doses; this reduces by a factor of 10 the activity of Kr-88 within about 12 h; Xe-138 has almost completely decayed within that interval.

(ii) The quick removal of aerosol particles from the accident atmosphere; this reduces by three to four orders of magnitude the 3.5 t mass of all the aerosols generated and released into the containment within the first twelve hours.

Taking into account prevailing uncertainties as regards the resuspension of radioactive aerosols from the sump water and from other surface condensates and with respect to the complex iodine chemistry in the radiation field, the following tentative values are proposed for filter loading (design basis values):

for elemental iodine = 5% of the core inventory

for aerosols = 60 kg (present loading capacity of the emergency standby filters)

Under the aspect of off-site emergency management the situation can be summarized as follows:

Using the set of parameters of

100% noble gas release into the containment,
95% iodine release as aerosols into the containment,
5% elemental iodine release into the containment,
1.5 Vol.%/d leak rate of the inner containment,
99% aerosol retention on filters, and

90% elemental iodine retention on filters,

the expected dose values remain far below the lower dose reference levels for initiating evacuation and relocation. If the aerosol filter efficiency is raised from 99% to 99.9%, the safety margin increases by roughly one order of magnitude. An increase in iodine filter efficiency from 90% to 99% which reduces accordingly the release of gaseous iodine into the environment produces a similar effect. In the case of unfiltered releases, however, the safety margin between the lower reference levels for intervention and the expected doses is so small, that severe emergency actions such as evacuation and relocation cannot be excluded.

Generally, it can be concluded that a double containment with exhaust from the annulus of substances leaking from the inner containment likewise covers the uncertainties resulting from the present application of partly simplified models used to describe the fraction and the behavior of the radionuclides in the containment atmosphere – provided that the accident filters are appropriately designed.

REFERENCES

1. D. Queniart, G. Keßler
"Common safety approach for future pressurized water reactors in France and in Germany"
Proc. of the Quadripartite Meeting of the National Advisory Committees on "Safety Options for future pressurized-water reactors",
Luynes, France, Oct. 1993.

2. G. Keßler, H.H. Hennies, J. Eibl
"Severe accident containment loads and possible design concepts for future large PWR's"
to be published in Nuclear Technology.

3. B. Kuczera
"R&D activities on safety aspects of future PWR plants performed at KfK"
Nuclear Safety, Vol. 34, No. 2, April - June 1993, p. 213.

4. G. Keßler et al.
"On the confinement of the radiological source term during beyond design basis accident events in future pressurized water reactors"
to be published in Nuclear Technolgy.

5. G. Keßler et al.
"Zur Eingrenzung des radiologischen Quellterms bei auslegungsüberschreitenden Ereignissen in zukünftigen Druckwasserreaktoren"
Report KfK-5199, August 1993.

6. F. Gelbard
"MAEROS users manual"
Report NUREG/CR-1391, 1982.

7. K.K. Murata et al.
"Users' Manual for CONTAIN 1.1 - A Computer Code for Severe Nuclear Reactor Accident Containment Analysis" NUREG/CR-5026, SAND87-2309, (Nov. 1989).

8. COSYMA: A New Program Package for Accident Consequence Assessment.
Joint Report by Kernforschungszentrum Karlsruhe GmbH and National Radiological Protection Board
Commission of the European Communities, Report EUR-13028 (1991).

9. COSYMA: User guide.
Compiled by I. Hasemann and J.A. Jones
Commission of the European Communities, Report EUR-13045 EN (1991)
Karlsruhe, Report KfK-4331 B (1991)

10. H.-J. Panitz, C. Matzerath, J. Päsler-Sauer
UFOMOD: Atmospheric Dispersion and Deposition
KfK-Bericht 4332 (1989).