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AUG 9 1985

MEMORANDUM FOR: J. McKnight
Records Services Branch, TIDC/ADM

FROM: C. Ryder
Fuel Systems Research Branch, DAE

SUBJECT: DOCUMENTS FOR THE PUBLIC DOCUMENT ROOM

Enclosed are reports that supplement NUREG-0956, "Reassessment of the Technical Basis for Estimating Source Terms." These reports are in addition to those sent with my memorandum dated August 10, 1985. I would like these reports placed in the Public Document Room.

C Ryder
C. Ryder
Fuel Systems Research Branch
Division of Accident Evaluation

Enclosure: As stated

"GESSAR-II, BWR/6 Standard Plant Probabilistic Risk Assessment General Electric Company 1982"

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CURRENT RESULTS OF RADIOACTIVE SOURCE TERM ANALYSES FOR MELT DOWN SEQUENCES IN KWU-TYPE PWR's

K. Hassmann, M. Fischer

Kraftwerk Union AG, P.O.Box 3220,
D-8520 Erlangen, F.R.G

J. P. Hosemann, H. Bunz

Karlsruhe, Nuclear Research Center,
P.O.Box 3640, D-7500 Karlsruhe, F.R.G

ABSTRACT

Core melt accidents in nuclear power plants are analysed to occur with an extremely low probability. In spite of this fact the public interest concentrates on the radioactive material which may be released during these highly improbable hypothetical events. Therefore, the objective of the R+D-work sponsored by the Federal Ministry of Research and Technology focus on the development of the analytical basis for radioactive source term predictions. It is the purpose of this paper - on the basis of 1300 MWe KWU type PWR's - to summarize and to compare the results of current analyses performed in the Federal Republic of Germany.

INTRODUCTION

The investigations into core meltdowns conducted for more than ten years in the Federal Republic of Germany (FRG) currently have been concentrated on the source term for risk dominant accident sequences. To determine the radioactive fission product releases to the environment, thermohydraulics and thermodynamics are needed as well as comprehensive computations to evaluate the aerosol and iodine fractions and their physical and chemical behavior. Additionally, single problems - such as the pressure level beyond that the steel shell fails during pressurization as well as the flow pathes of the gases within the reactor building before and after containment failure - influence the results and therefore have to be identified. Recent analyses performed currently demonstrate that the consequences of melt downs had been conserably overestimated in previous basic studies such as the German Risk Study / 1 /. The above statement is based not the least on the particular features of the containment and the entire building with respect to the retention of radioactivity. A typical German containment is shown in Fig. 1.

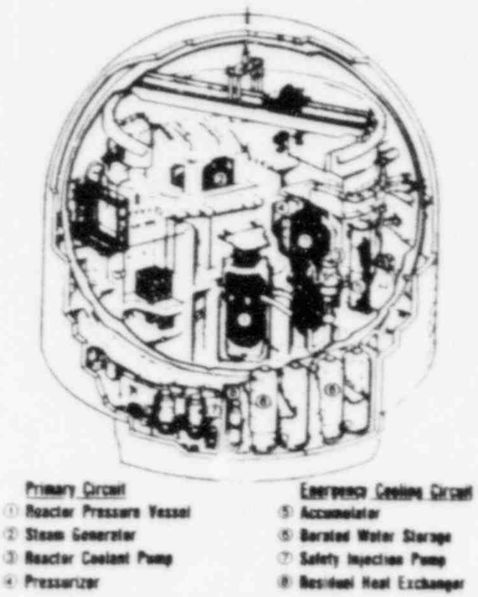

Fig. 1: **Reactor Building Biblis**

ACCIDENT SEQUENCES

With a view of the considerations to be made later in this paper, it is appropriate to present first a review of the melt-down sequence. In general, two cases can be distinguished as typical examples: the low pressure and the high pressure paths which with respect to the radioactive source term are expected to be the enveloping bounds for all other melt-down sequences.

Low Pressure Path

After a large leak as initiating event this sequence proceeds at low pressure in the primary system. Representative of this category, the sequence will be described following a double-ended break of the hot main coolant line and assuming the failure of the low pressure emergency core cooling system if the operation changes from the feed to the sump recirculation mode. As a consequence of this hypothesis the evaporation from the reactor pressure vessel flooded up to the main coolant line level starts 20 minutes after blow-down. Subsequently the water level after 0.6 h has dropped down to the upper edge of the core followed by failure of the core support structure after another 1.2 h.

The interaction of core melt with the foundation concrete starts immediately after failure of the reactor pressure vessel about 1.9 h after blow-down. Evaporation of the sump-water starts after about eight hours, immediately after the concrete shielding in the reactor cavity is penetrated which, initially, keeps the sump separate from the melt. The long-term build-up of pressure in the containment is determined by the evaporation of the sump-water. With the containment isolated - which exhibits only the design leakage of 0.25 Vol%/d - this gives rise to a pressure build-up in the long run.

High Pressure Path

Contrary to the low pressure path, this sequence is characterized by events taking place at high pressure in the primary system. If, after an emergency power case additionally the whole set of redundant Diesel generators fails, no electricity supply is available. In that case, the decay heat is initially removed from the core to the steam generators which, on the secondary side, evaporate their water inventory until after about 1.5 h all steam generators are getting dry. This results in a pressure and temperature rise in the primary circuit onto the set point of the pressurizer pressure relief valves.

Closing and opening cycles of the pressure relief valves are repeated until the dropping water level has rendered bare parts of the core in the reactor pressure vessel. The core is further heated and later there will be indications of melting. If the reactor pressure vessel fails due to contact of molten material with the RPV, substantial energy and mass transport takes place from the primary system into the containment. Depressurization upon failure of the reactor pressure vessel is followed by flooding of the molten material and of the still unmolten core parts slumped on the concrete foundation through accumulator water. Depending on whether the liquid and solid core parts are coolable or not coolable in water, the water evaporates at a slightly faster or slower rate. For both, the low and the high pressure cases Table 1 compares the most important results of the sequence analyses. The lowest three lines contain important information which is needed to value the fission product release rates presented in the following sections.

In conformity with the results reported in the German Risk Study on Nuclear Power Plants it can be assumed that in much more than 90 % of all conceivable cases the containment is tight at the onset of an accident (except for the design leakage to be considered) and that it remains tight until overpressure failure. Therefore, the calculations presented in the paper concentrate on this scenario.

	Low Pressure Case	High Pressure Case
Initiating event, h	0.3	0
Core heat up, failure of core support structure after, h	1.3	5.0
RPV-heat up, until h	1.9	5.0
Sump water ingression, h	8	5.0
integral aerosol release into containment, kg	3460	22.4
Failure of steel shell, d	5	Melt coolable yes: 4.3 no: 5
Sump evaporated, d	12/8.5*	6/7*

* depending upon failure mode of the steel shell (20/300 cm^2)

Table 1: RESULTS OF MELT DOWN SEQUENCES

THERMODYNAMICS IN THE CONTAINMENT AND IN ADJACENT VOLUMES

The calculations have been performed with the computer code WAVCO / 2 / which has been developed in order to predict the thermodynamics and the distribution of gases and other constituents within a subdivided building. Based upon the main features illustrated in Fig. 2, an equation system consisting of separate mass- and energy-balances for the state of the atmosphere and sump of each zone is set-up. Furtheron, additional balances for the mass of each component must be solved to determine the actual gas distribution. All possible thermodynamic states of the atmospheric steam are commonly covered by the same equation-system. Since the conditions inside different zones are strongly dependent from each other all the zone-specific equations have to be combined to form a coupled non-linear differential equation system.

Fig. 3 illustrates the overall reactor building as well as the flow paths which have to be considered in core melt situations of release category 6 sequences (containment intact within the first couple of days until overpressurization of the steel shell). Fission products approaching the annulus can be released to the environment through the air extraction system via the stack - or - at slight overpressure conditions - via leakages through penetrations in the outermost concrete structure. After containment failure the pressure in the annulus increases beyond the 0.1 bar limit. Then, a large leak with an area of 12 m^3 which has been identified to be the weakest point fails resulting in a pressure equalisation between the annulus and the auxiliary building. During this blow-down procedure the pressure in the auxiliary building increases up to the 0.05 bar level initiating failure of a connection to the environment and to the turbine hall.

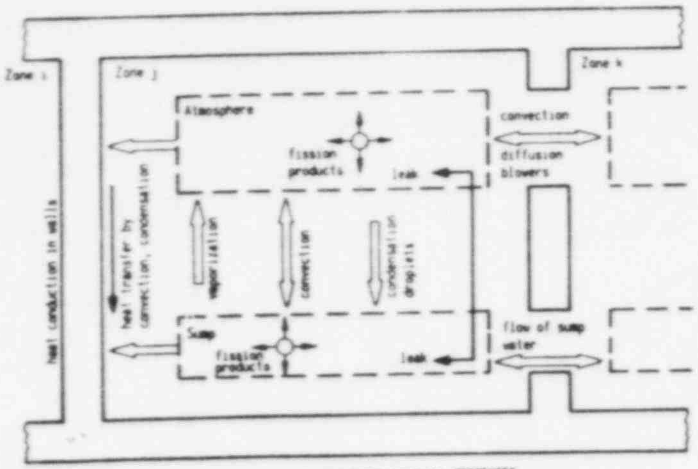

Fig. 2: WAVCO, SCHEME OF CONSIDERED PHYSICAL PROCESSES

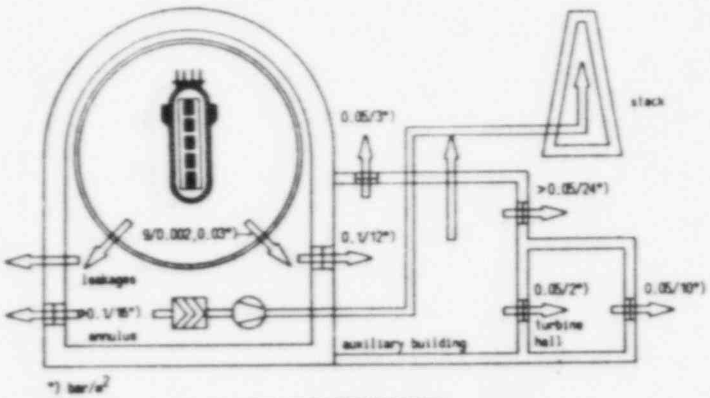

Fig. 3: SKETCH OF THE REACTOR BUILDING

From that point on, containment atmosphere enters the environment through several connections. The time dependent flow rates via the different flow paths has been calculated and have been used as the basis for aerosol and iodine calculations.

The containment failure mode, in particular, the cross section, influences the flow rate out of the containment into adjacent areas and therefore determines the released radioactive source term. As a matter of fact, experts in the FRG agree today that failure of the undisturbed steel shell of a German 1300 MWe Standard PWR will occur at about 14 bars and that in the realistic case at a lower pressure level leaks [illegible] before failure are expected. Therefore, investigations of the load carrying capability were performed at different locations of the containment, the goal being to quantify the type of failure for a steadily rising internal pressure and to indicate the associated cross sections of the openings.

At a pressure of 11 bar and a temperature of 170 [illegible] expansions of as much as approx. 40 cm and vertical tangential displacements at the equator of about 30 cm occur in the undisturbed shell zone. Deformations of this size are not [illegible] ed by the surrounding structure; even before attaining the loading condition indicated before, substantial constrained deformations take place at the disturbed points and hence leakages develop. The results show that failure of the steel shell must be expected to occur first at the material lock which is bolted to the steel shell of the reactor containment. For verification a cheap experiment will be performed / 3 /, the results of which could be transferred directly to real conditions without requiring to develop and run an expensive computer program.

According to the present state of knowledge a leak of limited size is expected. The leak size ranges between [illegible] and a value which is sufficiently high to prevent a further continous pressure rise in the reactor containment. This [illegible] depends exclusively on thermodynamic parameters because just the energy and mass flows generated in the containment at the time of overpressure failure must be removed through the leak. The leak is also strongly influenced by the layout of the containment. For containments of German standard PWR's a [illegible] of cross section is sufficient to prevent a further pressure increase in the containment.

On the basis of the energy and mass release into the containment and the two limit cases encountered for overpressure failure (20 cm² leak and 300 cm² leak) Fig. 4 shows the pressure plot for the low (LPC) and the high pressure case (HPC). For the LPC a pressure increase up to the 9 bar load limit of the steel shell must be expected after about five days only. Because of the fact the core material has been assumed to be coolable, a slightly shorter time interval of 4.3 days has been calculated for the HPC.

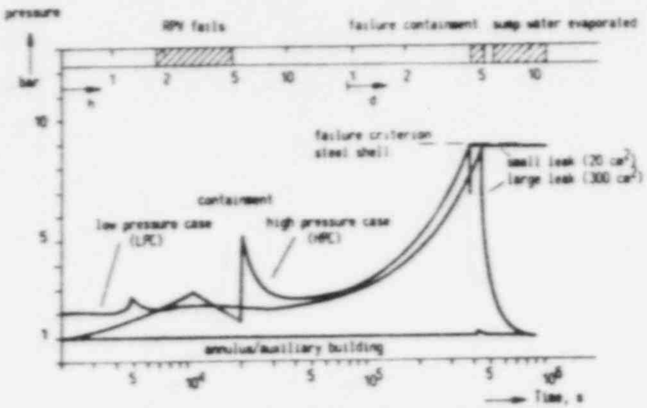

Fig. 4: CONTAINMENT HISTORIES FOR RELEASE CATEGORY 6 SEQUENCES

For the HPC the maximum pressure occuring during pressurization of the primary circuit is well below the design pressure of the containment. As already mentioned, the pressure can be stabilized with a 20 cm² leak whilst a leak cross section of 300 cm² leads to complete depressurization. The pressure in the annulus and the auxiliary building stays at the atmospheric level. Only immediately after failure of the steel shell the pressure within the annulus exceed to the 0.1 bar level causing failure of the connection between annulus and auxiliary building. This fact has been predicted to occur for both the small and the large leak size.

FISSION PRODUCT BEHAVIOR

Until overpressurization of the steel shell only the design leakage from the containment into the annulus is effective (about 7 m³/h). In the low pressure case fission gases, iodine and particles are transported to the stack passing the filter system via the annulus suction system (about 600 m³/h). Because of the loss of power this system doesn't operate in the high pressure sequence. For all cases the filter itself has been assumed to be ineffective at the time of overpressurization failure. This is a very conservative assumption because the situation within the annulus during and after this time period will probably only cause a degradation of the filter's behavior to retain iodine and aerosols.

Aerosols

Data derived from the SASCHA experiments / 4 / have been used in order to compute the release of aerosols particles from the core and the primary circuit into the containment. The behavior of the aerosol system within the containment and the adjacent volumes have been analysed by the NAUA-Code / 5 /. The code is based on physical aerosol processes summarized in Tab. 2 which also includes the sensitivity of each individual process on the basis of the conditions typical for LWR-scenarios.

Aerosol Process	Integrated in NAUA	Sensitivity
sedimeniation	yes	very important
diffusion	yes	minor effective
thermophoresis	no	insignificant for LWR-szenarios
diffusiophoresis	yes	important
turbulence	no	not important for LWR-szenarios
agglomeration	yes	very important
steam condensation	yes	important, if thermodynamics available

Table 2: SENSITIVITY OF DIFFERENT AEROSOL DEPLETION MECHANISMS

For the high and the low pressure case Fig. 5 shows the instantaneous airborne particle mass in the containment.

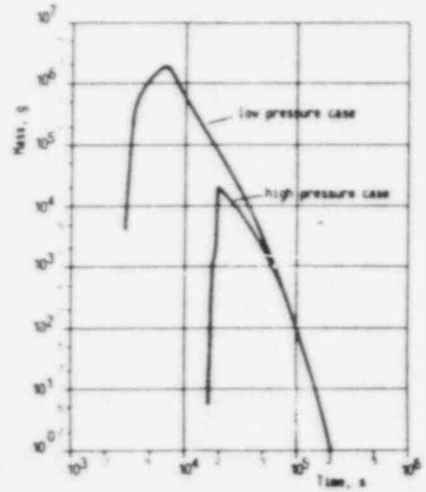

Fig. 5: AIRBORNE AEROSOL MASS IN THE CONTAINMENT

It should be pointed out that, obviously, by far the highest amount of airborne particle mass are non-radioactive elements and isotops. For the low pressure case (LPC) as a result of the large aerosol source the airborne mass decreases by more than five orders of magnitude within five days through aerosol-physical removal mechanisms. At the time of containment failure only those substances can still be released at the maximum which continue to be airborne. Starting with a less dense aerosol atmosphere the removal in the high pressure sequence (HPC) is slower and in the long term period equals the situation in the LPC. At the time of overpressurization of the containment which is not shown in Fig. 5 even higher aerosol concentrations are calculated for the HPC.

Fig. 6 shows the integral particle mass transported to the environment via all open connections as calculated on the basis of the flow paths illustrated in Fig. 3.

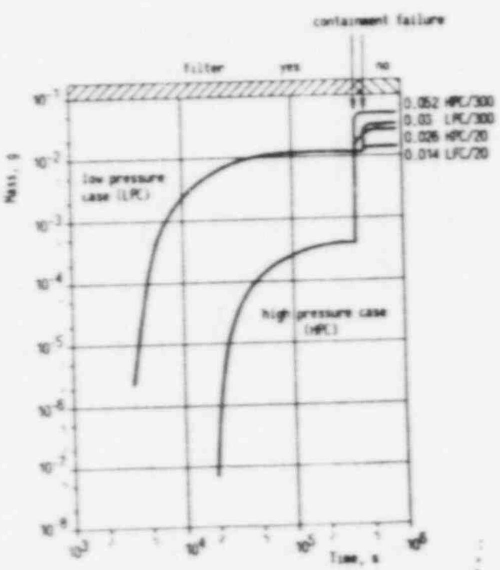

Fig. 6: ACCUMULATED AEROSOL MASSES LEAKED TO THE ENVIRONMENT

Compared with the small leak cases the results indicate about a factor of 2 higher releases for the 300 cm² leak. As a consequence of the shorter time interval until containment failure and of the higher amount of aerosol still airborne a slightly larger mass will be released to the environment in the HPC. It also can be concluded that failure of the filters as pessimistically assumed is more sensitive in the HPC.

Iodine

The iodine behavior within the containment and adjacent volumes has been calculated using the iodine model IMPAIR. The main features of the model / 6 / are summarized in Tab. 3.

During Release out of the Fuel and within the Primary System

- Constant release rates and homogenious mixing
- 99 % of I_2 instantaneously reacts in $Cs/H_2O/H_2$ atmosphere to form CsI
- no AgI reaction
- no retention
- in the high pressure case max. concentration of aerosols 200 g/m³ containing 2 % I

In the containment

- I_2-partition coefficient 200
- secondary reactions neglected which result in higher pH-values as f.i. impurities, IO_3-formation, Redox-potential, radiolysis
- I_2 reacts in sumpwater to form AgI, equilibrium after 3.5 h with 10 % of the I_2 available
- airborne I_2 reacts with organic material to form organic I, at the maximum 90 % I_2 and 10 % organic I

Table 3: IMPAIR (IODINE MODEL), IMPORTANT ASSUMPTIONS

In Adjacent Volumes

- carry over of I-species corr. to mean ... and composition within the source-volume
- no I⁻ release to adjacent volumes via water droplets in the steam flow generated by sump water evaporation
- partition coefficient of 5000 for the overall I
- organic I released to adjacent volumes doesn't react any more. Additional organic I is formed by airborne I_2 (after 10 h 50 % of each)
- I-release continues until all the sump water is evaporated

Tab. 3 cont.: IMPAIR (IODINE MODEL), IMPORTANT ASSUMPTIONS

These assumptions are based on the actual knowledge in this extremely complex science and represent the commonly agreed opinion of experts in the FRG and abroad. Nevertheless, important assumptions which usually include some conservatism need final experimental verification. The numbers which are reported in this paper will change in the future and therefore should be taken as a preliminary orientation.

Analogous to Fig. 6 representing the aerosol release rates Fig. 7 shows the total iodine releases to the environment for the low and the high pressure case (HPC). Because of the uncertainties as well as the only small differences in the calculated results the influence of the different containment failure modes on the iodine behavior is not presented. Similar to the results of the NAUA calculations, larger iodine releases - at the end in the one/two orders of magnitude range - have been analysed for the HPC.

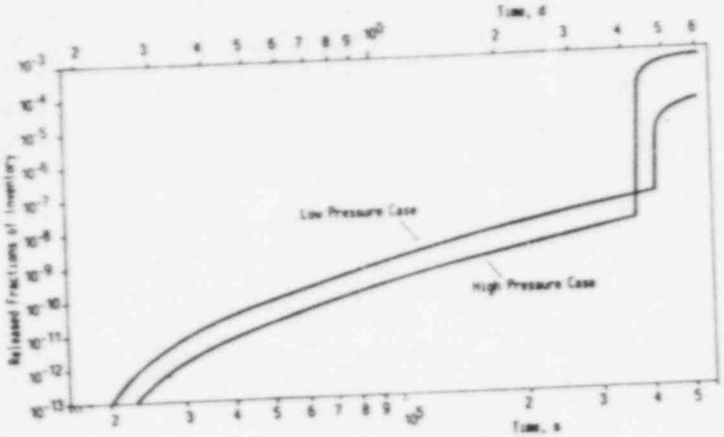

Fig. 7: COMPARISON OF THE TOTAL IODINE RELEASE TO THE ENVIRONMENT

On the basis of the HPC the fraction of the different iodine species is given in Fig. 8.

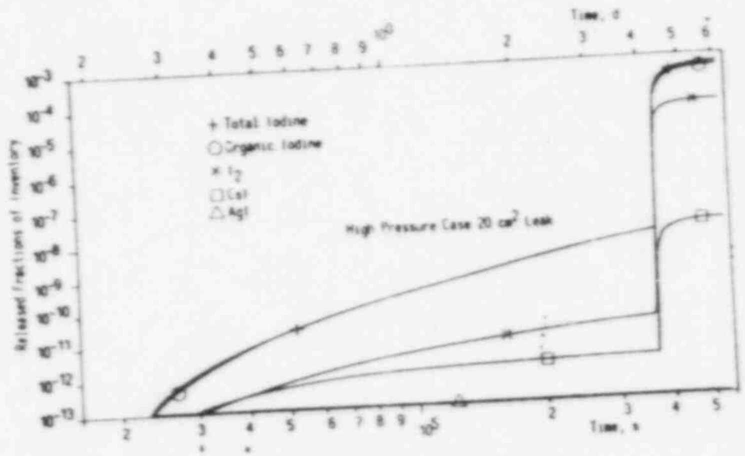

Fig. 8: IODINE RELEASE TO THE ENVIRONMENT

This tendency demonstrating the importance of organic iodine and I_2 has been identified for all scenarios.

CONCLUSION

Finally, Table 4 summarizes the integral release of Cs 137 and I 131 as the isotopes dominating the melt-down consequences, I 131 determining the early fatalities, Cs 137 the late cancers and the ground contamination. Although care should be taken as a consequence of important assumptions which may change in the future, it can be concluded that - compared with the results of the German Risk Study - the consequences of core melt-downs will be much lower than previously estimated.

	Cs 137		I integral	I 131	
	q	C_I	q	q	C_I
Low Pressure Case - case 1 - case 2	$3 \cdot 10^{-4}$ $7 \cdot 10^{-4}$	$3 \cdot 10^{-2}$ $7 \cdot 10^{-2}$	1	0.04	$5 \cdot 10^{3}$
High Pressure Case - case 1 - case 2	$3 \cdot 10^{-3}$ $5 \cdot 10^{-3}$	0.3 0.5	10	0.5	$6 \cdot 10^{4}$

case 1 = 20 cm² leak
case 2 = 300 cm² leak

Table 4: CAESIUM AND IODINE RELEASE TO THE ENVIRONMENT (WITHOUT DECAY)

To confirm the expected tendency ongoing research work on specific aspects related to severe accidents has to be completed. In particular, this includes: sensitivity studies, consequence analyses for other dominant sequences, improvements of fission products behavior and demonstration of aerosol plate-out (DEMONA-experiments). In addition, work is being performed to clarify hydrogen distribution and explosion phenomena as well as the long term melt/concrete behavior (BETA-experiments). All the R & D work mentioned above has been initiated and is expected to be completed in the near term future.

LITERATURE

/ 1 / "Deutsche Risikostudie Kernkraftwerke", Herausgeber Bundesminister für Forschung und Technologie, Verlag TÜV Rheinland, Köln 1979

/ 2 / M. Fischer et al, The Multi Compartment Code WAVCO, Status and Validation, International Conference on Containment Design, June 17 - 20, 1984, Toronto, Ontario

/ 3 / H. H. Hennies et al, The Reactor Containment of German Pressurized Water Reactors of Standard Design, as / 2 /

/ 4 / H. Albrecht, H. Wild, "Behavior of I, Cs, Te, Ba, Ag, In, and Cd during Release from Overheated PWR Cores", International Meeting on Light Water Reactor Severe Accident Evaluation, Aug. 29 - Sept. 1, 1983, Cambridge, Mass.

/ 5 / H. Bunz, W. Schikarski, W. Schöck, The Role of Aerosol Behavior in LWR Core Melt Accidents, Nuclear Technology, Vol. 53, pp. 141-146, May 1981

/ 6 / J. P. Hosemann, K. H. Neeb, J. Wilhelm, Neuere Bewertung der Freisetzung und des Transportes von Jod bei schweren Reaktorstörfällen, Jahrestagung Kerntechnik 1983, Berlin