IN13-mf--3070

DE9000069

COMPUTER CODES DEVELOPED IN FRG TO ANALYSE HYPOTHETICAL MELTDOWN ACCIDENTS

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1. Introduction

It is the purpose of this paper to give the status of significant computer codes developed within the Project "Core Meltdown" of the Light-Water-Reactor Safety-Research Program funded by the Federal Ministry of Research and Technology (BMFT). The Project Core Meltdown was initiated as basic research and development program to provide information for LWR-risk analyses due to hypothetical meltdown accidents. The specific objectives of the analytical investigations are given in Fig. 1.

Individual computer codes were developed by different contractors listed in Fig. 2 and by the Kernforschungs-Zentrum Karlsruhe (KFK, Fig. 3) to determine the course of the hypothetical melt down accident. The accident was subdivided into four phases; heat up of the core (1st phase), evaporation of residual water (2nd phase), heat up of reactor pressure vessel (RPV) (3rd phase) and penetration of the concrete foundation (4th phase). In addition, computer codes are available to analyse pressure and temperature in the containment atmosphere. The four phases are shown schematically in Fig. 4. The names of the codes available for investigation of all four accident phases and containment response are given in Fig. 5.

2. 1st phase: Core heat up until failure of the core support structure

Two different sophisticated codes are available to describe the course of the first phase (Fig. 5). In MELSIM (RS 73 in Fig. 2) the core, as well as its surrounding structure, is modelled in r-z-geometry for both types of LWR. The code is composed by a number of different modules, each describing a group of physical sequences during the first phase. By example, for a Boiling Water Reactor (BWR) the propagation of the slumping isotherme (1900 C) and lowering of the upper part of the core calculated by MELSIM is shown in Fig. 6. Starting with a fully dry core immediately after blow down the fuel pins start melting after 700 s. Because of the assymmetric axial and radial power profile slumping occurs outside the centre line especially in the lower part of the core. After 1000 s considerable part of the core in the region around the axis has reached the slumping criterion (1900° C). Anyway at the radial boundary about 30 percent of the fuel elements are not damaged at this time.

In the KAUHZ-code (RS 72 b in Fig. 2) the reactor core of a PWR is represented by an average powered fuel pin which is subdivided into several axial zones. All decay heat produced in such fuel pins still covered by water will be assumed to evaporate the water. The temperature of the fuel pin above the water level will increase because the steam flowing off into the containment does not provide sufficient cooling. After the temperature of an axial pin zone has reached the slumping temperature of 1900° C [1], the mass of the zone drops in the residual water. A 30 % reduction of the total decay heat is assumed, to take into account release of gaseous and volatile fission products (PNS 4243 in Fig. 3). Fig. 7 shows the decrease of the water level and propagation of the melting front calculated by KAUHZ. Twenty minutes after blow down residual water in the reactor pressure vessel (RPV) is at the primary coolant pipe level. The upper grid plate level will be reached after approximately 40 min. After 1.65 hours all water in the core region will be evaporated. At that time, about 2/3 of the representative fuel pin is molten and the slumped fuel fragments may have damaged the core support structure.

3. 2nd phase: Evaporation of the residual water

The analyses of the 2nd phase were made by WAVER (RS 72 b in Fig. 2). In addition LUECKE (RS 211) has been developed by Uni-Stuttgart to calculate the evaporation of residual water in more detail. Calculations have shown that all the water, originally contained in the bottom head of the RPV, is evaporated 0.65 hours after damage of the core support structure.

4. 3rd phase: Heat up of the reactor pressure vessel

After evaporation of the residual water in the RPV, a pool of molten fuel, steel and zircaloy called CORIUM [2] will react with the pressure vessel steel. Heat transport from the melt to the vessel wall takes place predominatcly by natural convection. THEKAR (RS 48/1 in Fig. 2) has been developed to investigate the thermohydraulic process within the molten mass. Fig. 8 shows in the upper part the calculated streamlines and isotherms, characterizing the movement of the fluid in the spherical calotte of the RPV. The numerical values shown are dimensionless. The heat transfer represented by relative Nusselt numbers over angle is given in the lower part of Fig. 8. The Nusselt number decreases drastically with the distance from the upper surface of the melt. As shown, the Ra-number which is a dimensionless grouping characterising the natural convection influences the heat transfer considerably.

Starting with heat fluxes at the outer layer of the melt calculated by THEKAR, the temperature distribution was analysed at the melt-covered and the dry parts of the pressure vessel with the computer code RAUHZ (RS 72 b in Fig. 2). Fig. 9 shows the temperature in the outermost layer of the RPV-wall covered by the melt as a function of time. After approximately 15 minutes, the temperature of the outermost layer of the pressure vessel has reached 1400° C at an axial location near the upper surface of the melt. A temperature of 1400° C is considered as the failure criterion for the pressure vessel, since metallurgical reactions were not found to substantially affect the melting point of steel [2].

5. 4th phase: Destruction of concrete

The bottom of the reactor pressure vessel will fail and the melt will come into contact with the concrete floor of the reactor cavity. Up to now, calculations of the penetration shape and the concrete destruction rate were made with the computer code BETZ (RS 183 in Fig. 2). The theoretical model

is based upon results of experimental investigation [3].
BETZ contains an energy- and mass balance for the molten pool and for the concrete. Because the heat conductivity of the concrete is low, BETZ does not take into account the heat transport into the concrete by conduction. However, the code KAVERN has been developed, which does consider heat conduction. In order to calculate the heat fluxes at the propagation front, two detailed thermohydraulic models have been established.
The BETON (RS 166 in Fig. 2) and the FILM-code (PNS 4244 in Fig. 3) determine the heat fluxes transferred to the concrete. The BETON-model assumes that heat is transported to the concrete by convection caused by rising single gas-bubbles. Fig. 10 shows the results computed by BETON in comparison with a temperature field that has been obtained in model experiments by holographic interferometry. The measured and calculated temperature fields show a thin termal boundary layer resulting in the large heat fluxes depicted in the lowest part of Fig. 10. The temperature of the melt in contact with concrete stays constant due to the mixing effect of the rising bubbles as can be seen by the low number of isothermes within the melt in the calculated and the holographic temperature fields. The FILM-code which is integrated in WECHSL (PNS 4331 in Fig. 3) calculates the heat transfer, assuming that a gas film will be generated between the molten pool and the concrete.

Fig. 11 shows the geometry of the cavity at different times as calculated by BETZ. The melt penetrates the concrete basement of a standard PWR with a thickness of approximately 6 meters in about 13.5 days. However a longer penetration time can be expected because the calculation in Fig. 11 doesn't take into account the flooding of the melt by sump water resulting in a considerably higher heat removal from the upper surface of the melt.

6. Pressure and temperature in the containment

Pressure and temperature history in the containment atmosphere can be analysed with the codes CONZU and COCO. Both models use a one compartment containment, taking into account all masses and energies entering the containment atmosphere during the four accident phases as shown in Fig. 12. The heat transport to the containment structures is also considered. Fig. 13 shows the pressure in the containment from the beginning of the evaporation of the residual water for 4 cases. These differ in treatment of the hydrogen combustion and sumpwater ingression. After 3.6 h a peak pressure of 6.5 bars will be reached if one assumes that combustion of the H_2 does not take place before a combustible H_2-air-H_2O mixture in the containment is formed. With continuous H_2-combustion the maximum pressure during the burning period will not exceed the maximum containment pressure of 4.8 bar occuring immediately after blow down. The pressure increase after 5 hours is caused by the assumption that the surface of the molten pool will be flooded by sump water. As a matter of this assumption high steam rates were generated which enter the containment atmosphere. After approximately 2 1/2 days the pressure reaches twice the peak value after blow down. Without sump water evaporation pressure increases not as rapidly due to the smaller mass and energy transport into containmentatmosphere. Starting with blow down on the top of the triangle, volumes of steam, air and hydrogen in the containment during the hypothetical melt down accident are shown in Fig. 14. Because of the sump water ingression, resulting in large steam contents in the containment, hydrogen production by melt-concrete interaction will not be sufficient to exceed detonation limits. With hydrogen combustion deflagration stops at 70 v/o steam and 30 v/o air content due to the fact that whole the oxygen has been burned.

7. Conclusions

Several sophisticated computer codes have been developed in the Federal Republic of Germany to analyse hypothetical melt down accidents for LWR's. It is planned to integrate all computer codes mentioned in Fig. 4 into an overall computer system based on the modular data-bases system RSYST /4/ in order to control the calculation and to manage the data transfer. The principal output of the code system includes the determination of time and duration of the different phases during the hypothetical core melt down accident as well as the pressure and temperature response of the containment including the fission product history.

Up to now calculations for melt down sequences with these codes have been developed under the hypothetical assumption that after a large LOCA the low pressure recirculation systems fail totally. The results summarized in Tab. 1 indicate that for a standard PWR, an overpressure failure of the containment is not expected to occur before 2 1/2 days. A more sophisticated analysis of the hypothetical core meltdown accident requires results of additional experimental and theoretical work which is under progress in the Federal Republic of Germany.

References

[1] S. Hagen, Experimental Investigation of the Meltdown Behaviour of LWR Fuel Elements, KFK-Nach., 7 (6): 45-49 (1975; also, Reactor Conference, Düsseldorf, 1976, AED-CONF-76-013-061, pp. 244-247)

[2] M. Peehs et al., Investigation of Metallurgical and Chemical Interaction between Core melt and RPV-Wall, FRGMRT-RS 74 A, Final Report, Kraftwerk Union AG, Erlangen, May 1976

[3] M. Peehs, A. Skokan, M. Reimann
Experimental Investigations in the FRG of the Barrier Effect of Reactor Concrete against Propagating Molten Corium in the case of a Hypothetical Core Melt Down Accident of a LWR. ENS/ANS Topical Meeting on Nuclear Power Reactor Safety, Brussels, Oct. 16-19, 1978

[4] R. Rühle,
RSYST I-III-Experience and Further Development
Atom Kernenergie Bd 26, 1975, S. 185-189.

Fig. 1

Objectives of the Analytical Investigations of the Project CORE MELTDOWN

- Define and Model Possible Core Meltdown Sequences
- Model Thermohydraulic Behaviour of Coremelts in Contact with Reactor Pressure Vessel, Concrete and Water
- Analyse Core Slumping and Formation of Molten Pool
- Complete Integral Code BILANZ and Integrate Coremelt Subroutines into this Code

Fig. 2 Most Important Analytical Investigations on CORE MELTDOWN of the BMFT Program Safety of Light Water Reactors

RS-No.	Subject
48/1 166	Thermohydraulic Behaviour of Molten Fuel in Contact with Reactor Pressure Vessel and Concrete (TU-Hannover)
72 a,b 183	Energy Balances in RPV, Containment and after RPV-Failure (KWU)
73 316	Development of MELSIM and Integration of Core Melt Subroutines into RSYST (UNI Stuttgart)
306 310 311	Best Estimate-ECC Calculations for Hypothetical Failure Combinations of ECC-Systems for PWRs (BBR, GRS, KWU)
PNS 4311 4331 4332 4333	KFK/PNS Programes see Fig. 3

PNS-No.	SUBJECTS	GOALS
4311	THE BEHAVIOR OF PARTICULATE RADIOACTIVITY IN THE CONDENSING STEAM ATMOSPHERE OF A POSTACCIDENT LWR-CONTAINMENT (BASED ON THE NAUA-EXPERIMENTS)	NAUA-CODE; QUANTITATIVE DESCRIPTION OF AIRBORNE RADIOACTIVITY AS FUNCTION OF TIME IN SAFETY ANALYSES
4333	RELEASE OF RADIOACTIVE AND NONACTIVE PARTICULATED MATTER FROM THE MELT INSIDE THE RPV AND FROM THE MELT INTERACTING WITH CONCRETE OF THE BASEMENT (BASED ON THE SASCHA EXPERIMENTS PNS 4334)	SOURCE-CODE; QUANTITATIVE DESCRIPTION OF THE PARTICLE RELEASE AND TRANSPORT AS FUNCTION OF THE TIME-TEMPERATURE HISTORY; SOURCE TERM FOR THE NAUA-CODE
4331, 4332, 4334	INTERACTION BETWEEN COREMELT AND THE CONCRETE OF THE BASEMENT, THERMAL DESTRUCTION AND EROSION OF THE CONCRETE, OXIDATION OF THE METALLIC PHASE, GAS RELEASE, HEAT TRANSFER MECHANISMS, DISTRIBUTION OF HEAT SOURCES INSIDE THE MELT (BASED ON THE BETA-EXPERIMENTS PNS 4323 AND ON THE SASCHA-EXPERIMENTS PNS 4334)	WECHSL-CODE;*) QUANTITATIVE DESCRIPTION OF THE PHENOMENA DURING THE MELT/CONCRETE INTERACTION-PHASE TO DETERMINE THE ALMOST POSSIBLE FAILURE OF THE CONTAINMENT BY GAS- AND STEAM OVERPRESSURE AND TO DETERMINE WHETHER THE MELT PENETRATES THE BASEMENT (WHEN ?) OR NOT *) DEVELOPED IN COOPERATION WITH SANDIA LABS./USA

Fig. 3 KfK PNS 4100/21.6.1978

CODE DEVELOPMENT AND THEORETICAL RESEARCH ON HYPOTHETICAL CORE MELTDOWN ACCIDENTS BY KFK/PNS

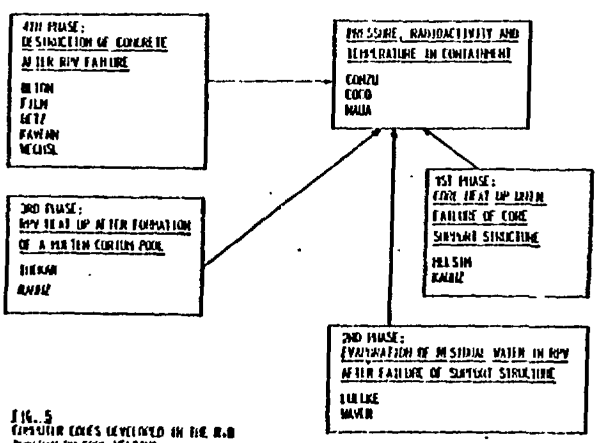


FIG. 5
COMPUTER CODES DEVELOPED IN THE R+D PROGRAM ON CORE-MELTING

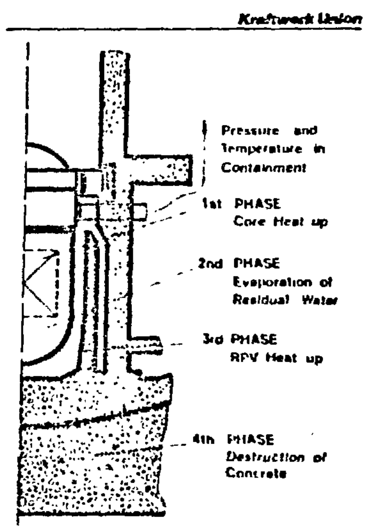


Fig. 4 The 4 Phases of Hypothetical Melt Down Accident

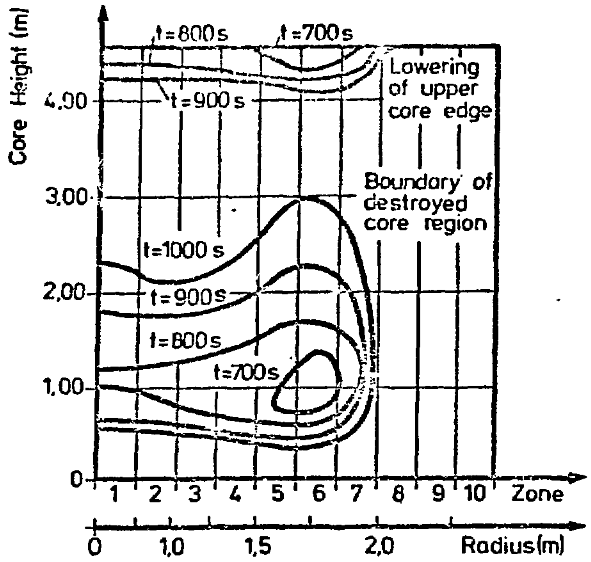
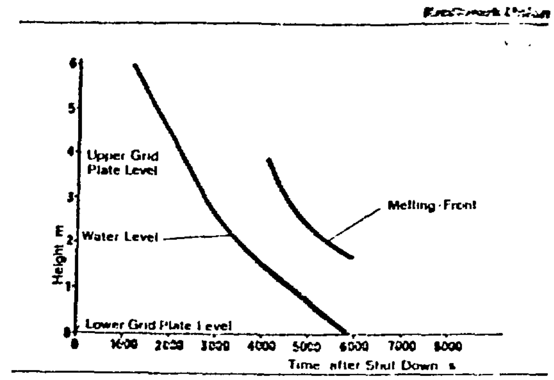


Fig. 6 Time dependent development of core slumping (BWR)

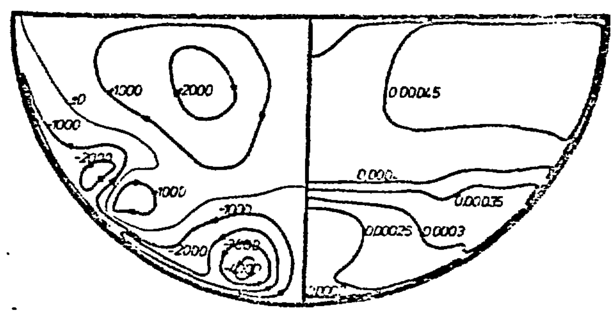


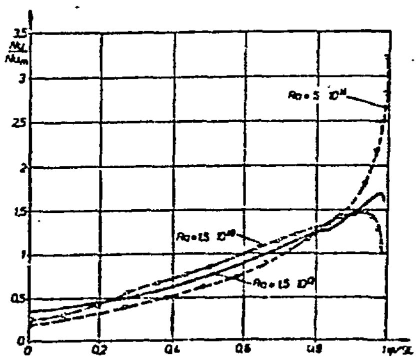
Decrease of Water Level and Propagation of Melting Front during Core-Heat-Up (1. Phase)

Fig. 7

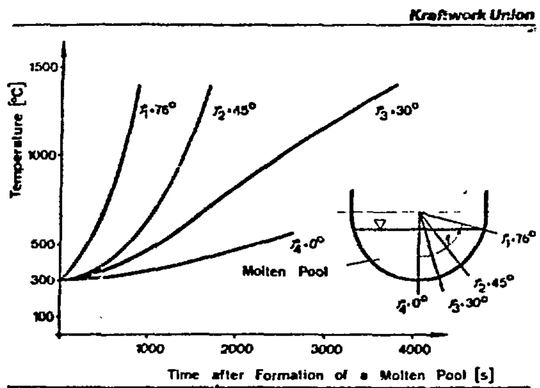
Fig. 8 *Thermohydraulic behavior and heat transfer in the corium at the reactor vessel bottom*

streamlines and isotherms



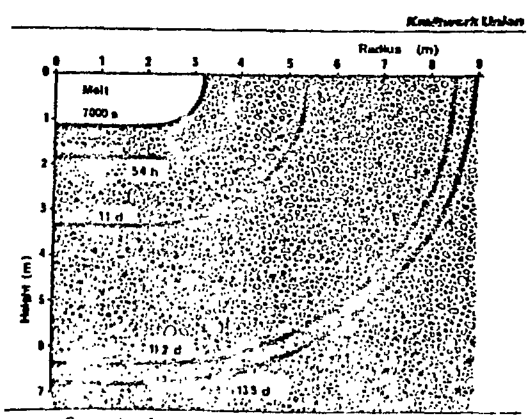


local Nu numbers



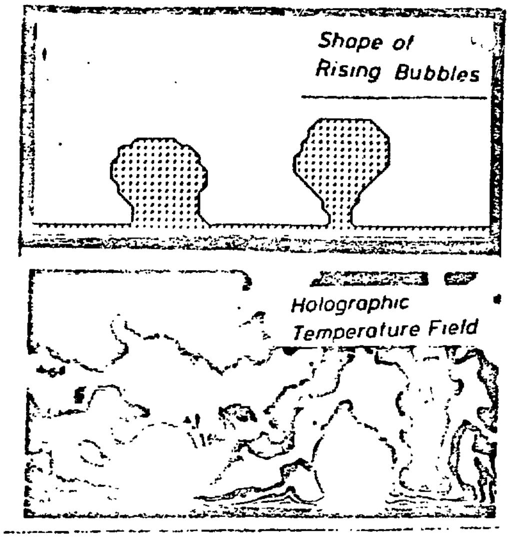
Temperature in Outermost Layer of Melt Covered RPV-Wall (3rd Phase)

Fig. 9

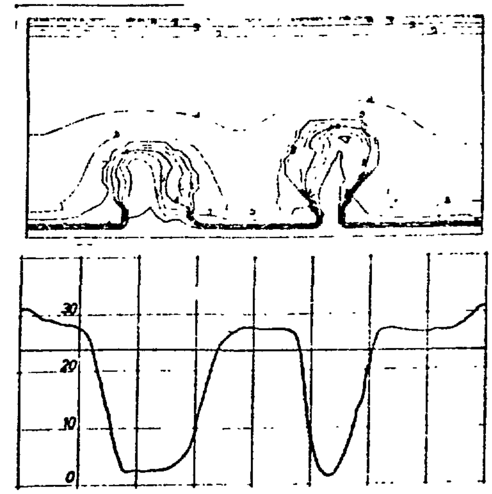


Concrete Destruction (4th Phase)

Fig. 10



Computed Temperature
Distribution Around
The Bubbles



Computed Heat Flux

Fig. 11 Computed Bubble Development
Code BETON 3.6

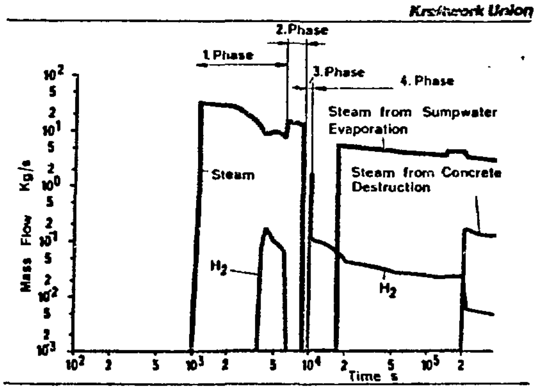
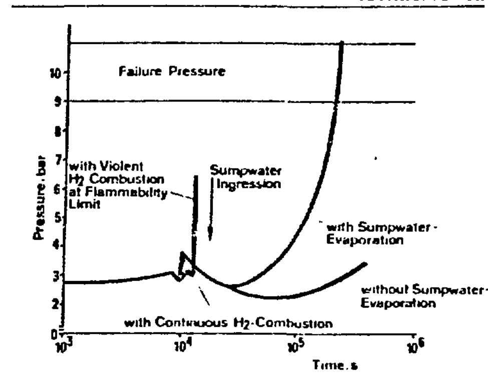
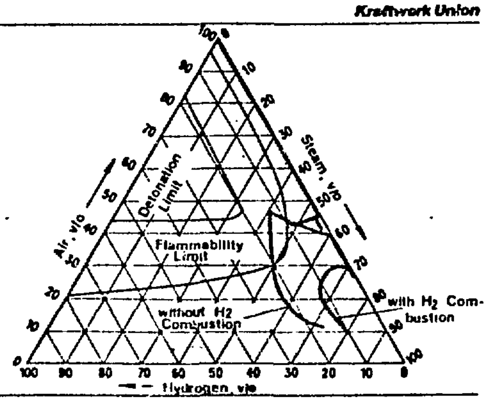


Fig. 12 Mass Transport into Containment



Pressure in Containment

Fig. 13



Flammability Limits of Hydrogen-Air-Steam Mixtures

Fig. 14

PHASE OF ACCIDENT	TIME FROM START OF ACCIDENT	COMPUTER CODE USED
EVAPORATION OF RESIDUAL WATER IN UPPER PLENUM STARTING AT	1150 S	
DURATION OF 1ST PHASE, UNTIL	1.65 H	KAIRZ
DURATION OF 2ND PHASE, UNTIL	2.3 H	WAVER
DURATION OF 3RD PHASE, UNTIL	2.45 H	KAUHZ
DURATION OF 4TH PHASE, UNTIL	> 10 [illegible]	BETZ
MAX. PRESSURE DURING CONTINUOUS H_2 COMBUSTION (2.8 H AFTER BLOW DOWN)	3.8 BAR	
MAX. PRESSURE DURING VIOLENT H_2 COMBUSTION (3.0 H AFTER BLOW DOWN)	6.5 BAR	COCO
FAILURE PRESSURE OF CONTAINMENT REACHED AFTER	2.5 D	

COMPARISON OF RESULTS

Tab. 1