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Severe Fuel Damage Investigations of KfK / PNS

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Projekt Nukleare Sicherheit

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K E R N F O R S C H U N G S Z E N T R U M   K A R L S R U H E

PROJEKT NUKLEARE SICHERHEIT

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Abstract

This report is a comprehensive review of the objectives, the program planning, the status and the further procedure of the investigations of KfK/PNS on severe core damage.

The investigations were started in 1981 and will be finished in 1985/86.

Untersuchungen des KfK/PNS zu schweren Kernschäden

Kurzfassung

Der vorliegende Bericht gibt einen kurzgefaßten Überblick über die Zielsetzung, das Versuchsprogramm, den derzeitigen Stand und das weitere Vorgehen bei den Untersuchungen des KfK/PNS zu schweren Kernschäden.

Diese Untersuchungen wurden im Jahre 1981 begonnen und sollen im wesentlichen 1985/86 abgeschlossen werden.

SEVERE FUEL DAMAGE INVESTIGATIONS OF KFK/PNS

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1. PROGRAM OBJECTIVE

Small break LOCA's and Special Transients which are postulated to occur in LWR's, in combination with loss of one of the required safety systems, can lead to overheating of the fuel rods and severe fuel damage without necessarily resulting in a core melt accident.

The objectives of the PNS Severe Fuel Damage studies are:

- to investigate the relevant physical and chemical phenomena which may lead to severe fuel damage,
- to develop models describing the extent of fuel damage if the cladding temperatures exceed the design limits and
- to quantify the safety margins presently existing in the safety systems of operating reactors and to explore their capability of ending a high temperature transient before it can lead to uncontrollable core meltdown.

The principal phenomena of interest are:

- the HT-Oxidation and embrittlement of Zry cladding and SS structural materials in steam at temperatures greater than 1200 °C, including possible steam starvation and hydrogen blanketing conditions,

- the resultant hydrogen production,

- the mechanical and chemical interaction between UO_2 and Zry cladding, especially the formation and behavior of liquified phases between Zry cladding and UO_2,

- the coolability of severely damage core geometries, and

- the fission product release from severely damaged cores.

2. PROGRAM DESCRIPTION

The overall program is divided into five major areas (see Fig. 1):

- Separate effects tests on:
  - HT oxidation of Zry and SS in steam
  - Interaction between Zry and UO_2
  - Equilibrium phase relations in the U-Zr-O system
- Out-of-pile single rod and bundle experiments with indirectly heated fuel rod simulators (CORA Program)
- Out-of-pile tests on the long term coolability of severely damaged core geometries (COLD Program)
- Annealing experiments on fission gas release at high temperatures
- Model and code development

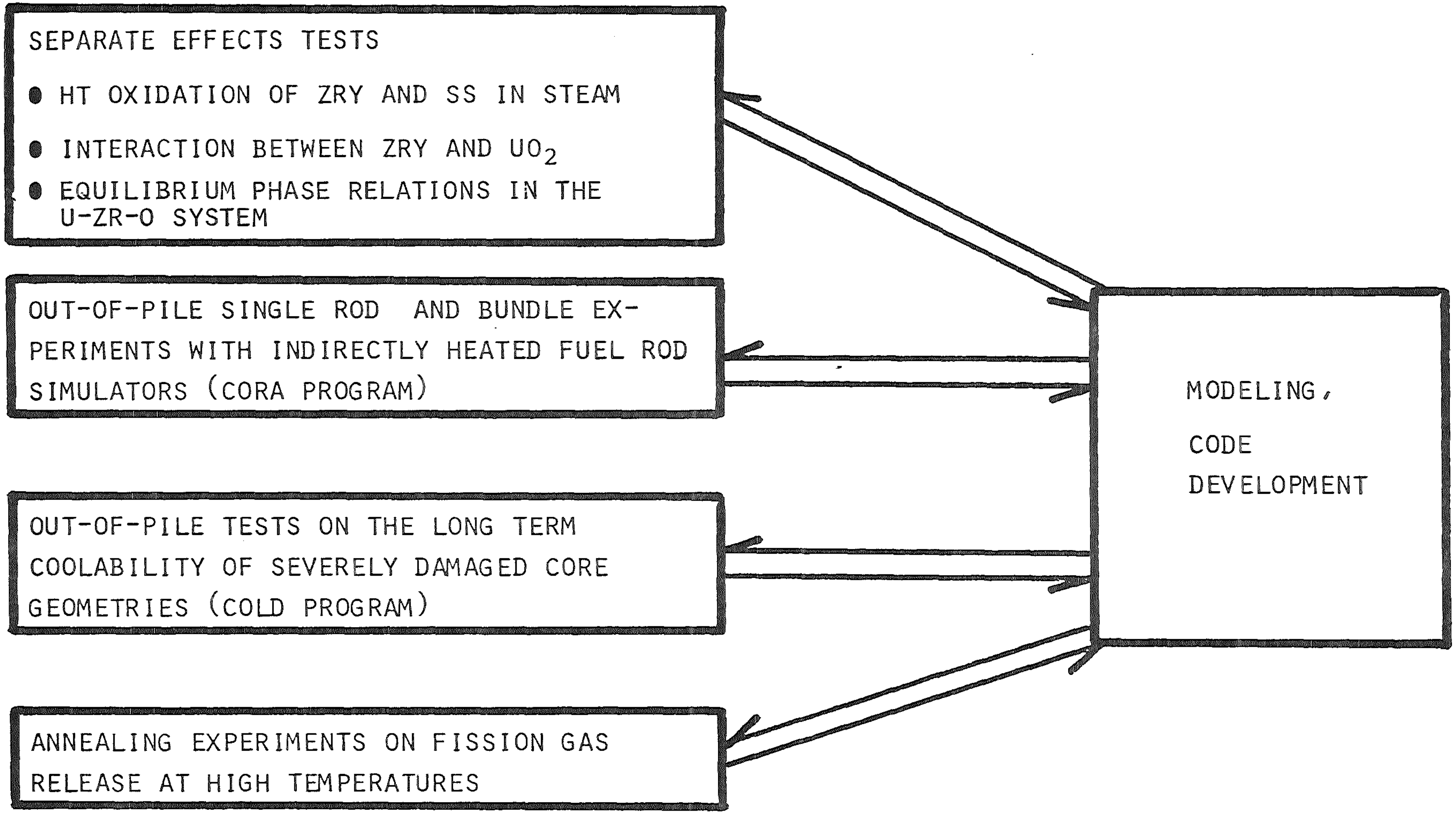


FIG. 1: SEVERE FUEL DAMAGE INVESTIGATIONS OF KFK/PNS (PNS 06.01.16)
PROGRAM SCHEME

2.1 SEPARATE EFFECTS TESTS

2.1.1 HT OXIDATION OF ZRY AND SS IN STEAM (S. Leistikow, IMF II)

TEST OBJECTIVES

As a contribution to the investigation of fuel rod behavior under Severe Core Damage Conditions the HT oxidation of Zircaloy 4 and Stainless Steels in steam and steam hydrogen mixtures and the mechanical response of the oxidized material are studied in separate effects tests. The objective of this program is to identify the relevant phenomena leading to and determining Severe Core Damage as a basis for the evaluation of integral tests and for code development.

PROGRAM PLAN

According to the objectives and the experimental equipment the program can be subdivided as follows:

1. Determination and interpretation of the HT oxidation kinetics of Zry 4 cladding in flowing steam

   1.1 Isothermal exposure beyond the LOCA relevant time/temperature range
      1.1.1 Reactions at higher temperature (< $1800^{o}C$)
      1.1.2 Reactions of extended duration (< 25 h)
         - Range of normal kinetics
         - Anomalies resulting from α/β-Zry phase transformation
         - ZrO_2 degradation and kinetical transition due to the breakaway effect

   1.2 Temperature transient exposure beyond LOCA relevant transients
      - Slow temperature ramps
      - Combined isothermal/transient determination of the breakaway existence regime
      - Special types of transients

2. Mechanical response of Zry 4 cladding to oxidation in steam

   2.1 Strength and ductility of pre-oxidized cladding
      - Protective ZrO_2 layers
      - non-protective (post-transitional) ZrO_2 scales

   2.2 Mechanical behavior during oxidation

   2.3 Oxidation behavior and mechanical properties of ballooned an fractured cladding

3. Investigation of the HT reaction kinetics of Zry 4 cladding in steam-hydrogen mixtures

   3.1 Determination of oxygen and hydrogen pick-up during isothermal exposure
      - Temperature and time dependence
      - Concentration dependence

   3.2 Dependence on temperature transient and stress/strain conditions

4. HT steam oxidation and mechanical behavior of austenitic and ferritic stainless steels

   4.1 Steam oxidation kinetics and microstructural response

   4.2 Burst testing of pre-corroded and oxidizing materials

TEST PROCEDURE

Steam oxidation kinetics of Zry 4 (program part 1):

The experiments are performed with 3o mm long tube sections in non-pressurized closed laboratory steam loops consisting of evaporator, superheater, ceramic test section and condenser. HT furnaces (1800^{o} C) and TC temperature control are used for the isothermal tests and direct induction heating (1500^{o} C) and optical pyrometer control for the transient tests. Test results are the oxygen uptake as well as the ZrO_2 and α-Zr(O) layer thicknesses and the microstructure according to the metallographic evaluation.

Mechanical behavior of oxidized Zry 4 (program part 2):

For tube rupture testing tube capsules of 5o mm length are connected with the argon system for internal pressurization (13o bar). For external oxidation the specimen is placed in an atmospheric steam loop within a HT furnace (1300^{o} C). Test results are burst pressure or time and circumferential strain in dependence of different test variables together with extent and morphology of oxidation.

Steam/hydrogen reaction kinetics of Zry 4 (program part 3):

The tests will be performed in a modified and not yet designed laboratory equipment, for which the control of three atmospheres (steam, hydrogen, inert gas) is an essential prerequisite. Dosage and control of H_2/H_2O proportion of the atmosphere as well as the analysis of H_2 and O_2 pick up of Zry 4 tubing are necessary for an understanding of the influences of hydrogen under less defined but more realistic conditions.

HT Steam oxidation and mechanical behavior of steels (program part 4):

Test facilities and procedure are in analogy to part 1 and to part 2.

PRESENT STATUS OF THE PROGRAM

Program part 1:

Oxidation tests at 135o-16oo^{o} C (1-6o min) with radiant and direct inductive heating are still in metallographic and kinetic evaluation and will be completed by suppelementary tests. Experiments towards the influence of the α/β-Zry transformation and the breakaway effect at 6oo-13oo^{o} C and extended duration (1-25 h) are essentially complete and the evaluation is continuing. Temperature transient testing will begin in the near future.

Program part 2:

Burst testing of pre- oxidized cladding is under way. After pre-corrosion for 1-35o min at 1ooo^{o} C internal pressures of 12-3o bar and continuing external oxidation are applied. Strength increase and reduction in ductility are compared to metallic specimen and inert gas conditions.

Program part 3:

The main test series will need preparation in equipment and evaluation methods. The time schedule is given in the attached table. Hydrogen uptake realized during long term steam oxidation under breakaway conditions is an indication of the interdependence with mechanisms of scale disintergration and is promising interesting results of the H_2 /H_2 O reaction program. In cooperation with KfK hydrogen considerations are being investigated at Soreq Nuclear Research Center, Yavne, Israel.

Program part 4

This work is performed in relation to a study of advanced LWR reactors using ss-clad fuel pins as well as to the behavior of structural materials of the classical LWR.

A steam oxidation kinetics test series with austenitic steel 15 % Ni-15 % Cr-1,3 % Mo-Ti-B (DIN 1.497o) in the temperature and time range 6oo-13oo^{o} C and 1o min to 6 h is completed and under evaluation. Similar tests are being performed with ferritic 12 % Cr steel (DIN 1.4914). Future work on rupture and ductility properties are projected in view of damage penetration far in advance of metal consumption for steam oxidized steels.

REFERENCES FOR CHAPTER 2.1.1

/1/ A.E. Aly: Oxidation of Zircaloy 4 Tubing in Steam at 135o to 16oo^{o}C. KfK 3358, Mai 1982.

/2/ S. Leistikow: Comparison of High Temperature Steam Oxidation Kinetics under LWR Accident Conditions: Zircaloy-4 versus Austenitic Stainless Steel. 6th International Conference on Zirconium in the Nuclear Industry, June 28 to July 1, 1982, Vancouver, BC, Canada.

Fig 2: HT Steam Oxidation of Zircaloy 4 under Severe Core Damage Conditions

Time Schedule

		1982	1983	1984
1. HT oxidation kinetics of Zry 4 in steam				
1.1 Isothermal exposure				
1.2 Temperature transient exposure				
2. Mechanical behavior and steam oxidation of Zry 4 cladding				
2.1 Preoxidized cladding				
2.2 Oxidizing cladding	1)			
2.3 Ballooned and fractured cladding	2)			
3. HT reaction kinetics of Zry 4 in H_2/H_2O mixtures				
3.1 Isothermal exposure				
3.2 "Realistic" conditions	3)			
4. HT steam oxidation and mechanical behavior of stainless steels				
4.1 Oxidation kinetics				
4.2 Mechanical testing				

1) Completed test series under evaluation

2) To be delayed or cancelled

3) To be projected if necessary

2.1.2 INTERACTIONS BETWEEN ZRY CLADDING AND OXIDE FUEL (P.Hofmann, IMF I)

TEST OBJECTIVES

- Investigation of the reaction kinetics between UO_2 and Zry-4 from 1000 to 2000 ^{o}C under inert and oxidizing conditions.

- Investigation of the influence of the mechanical interaction between the fuel and the oxidized cladding on the integrity of the fuel rod. Determination of the critical parameters which lead to fragmentation of the fuel rod.

PROGRAM PLAN

The program is divided into the following individual experiments series:

1. Examination of the UO_2/Zry-4 chemical interaction above 1500 ^{o}C under inert conditions and above 1000 ^{o}C under oxidizing conditions.

2. Determination of the UO_2/Zry-4 reaction kinetics and characterization of the reaction product (inert and oxidizing conditions).

3. Determination of the wetting behavior of UO_2 by molten Zry and molten α-Zr(O) (inert conditions).

4. Investigation of the mechanism of UO_2 dissolution by liquid Zry (inert conditions).

5. Determination of the UO_2 dissolution rate (inert conditions).

6. Determination of the maximum solubility of UO_2 in liquid Zry as a function of temperature, time, and oxygen content of the Zry (inert conditions).

7. Investigation of the effect of cladding oxidation on the UO_2/Zry mechanical interaction between 1000 and 1600 ^{o}C.

EXPERIMENT CONDUCT

Experiment series 1 and 2 (above) are performed in the high temperature/high pressure equipment MONA ($T_{max} \simeq 2000\ ^oC$, $p_{max} \simeq 200$ bar). The 10 cm long specimens are contained in a high pressure vessel and inductively heated, with the cladding serving as susceptor. Cladding temperature is monitored continuously with an optical pyrometer. The test temperatures range from 1000 to 2000 ^oC and the annealing time from 1 to 150 minutes. Heating rate, reaction time, and cooling rate are input to a computer and controlled automatically during the test. The external overpressure (to ensure good fuel/cladding contact) ranges from 2 to 80 bar.

The wettability of UO_2 by molten Zry or molten α-Zr(O) (series 3) is measured by the sessile drop method. A small cylindrical Zry-4 or α-Zr(O) specimen ($\sim$ 4 x 5 mm) is placed in the center of a polished UO_2 disc. The sample is inductively heated in argon with tungsten or graphite as susceptor. The test temperature ranges from 1800 to 2000 ^oC and is measured with a pyrometer. The test time ranges from a few seconds to one hour. Photographs are taken throughout each experiment to determine the change in wetting angle as a function of time at a specific temperature.

Experiment series 4, 5 and 6 are performed in argon in an induction furnace. A small amount of Zry-4 or UO_2/Zry mixture is placed in a UO_2 crucible. An outer tungsten crucible acts as susceptor. The test temperature ranges from 1800 to 2000 ^oC and is measured with a pyrometer. The experiments are interrupted after various reaction times (a few seconds to one hour) to determine the extent of the dissolution process.

Experiment series 7 is performed in a high temperature tube oven in steam.

STATUS OF THE WORK AND TEST SCHEDULE

First results of the UO_2/Zry chemical interaction experiments up to the melting point of Zry have been reported. Determination of the reaction kinetics is in progress. The wetting experiments are predominately complete. Evaluation of the test results has begun. The dissolution of UO_2 by molten Zry has been investigated. Experiments to determine the dissolution rate have not yet begun. First results of the effect of cladding oxidation on the UO_2/Zry mechanical interaction have been reported.

The present schedule for the performance and completion of the individual tasks is given in the attached table (Fig. 3).

REFERENCES FOR CHAPTER 2.1.2

/1/ P. Hofmann, C. Politis: The Kinetics of the Uranium Dioxide-Zircaloy Reactions. Journal of Nuclear Materials 87 (1979) 375-397.

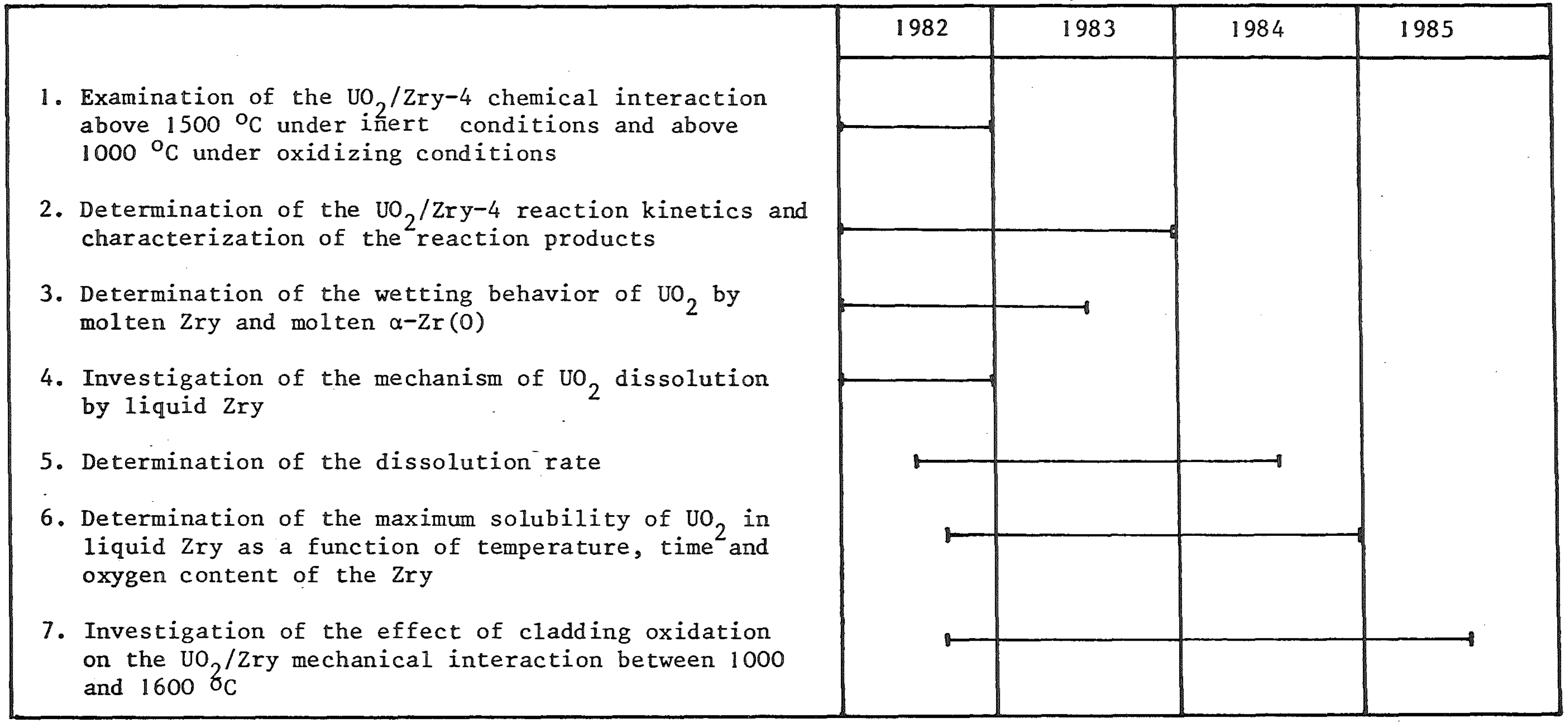
/2/ P. Hofmann, D. Kerwin: Preliminary results of UO_2/Zircaloy-4 experiments under Severe Fuel Damage conditions. IAEA Specialists' Meeting on "Water Reactor Fuel Element Performance Computer Modeling", Preston, England, March 14 - 19 (1982).

/3/ P. Hofmann, D. Kerwin-Peck, P. Nikolopoulos: Physical and chemical phenomena associated with the dissolution of solid UO_2 by molten Zircaloy-4. 6th International Conference on "Zirconium in the Nuclear Industry", Vancouver, Canada, 28 June - 2 July (1982).

/4/ W. Dienst, P. Hofmann, D. Kerwin-Peck:
Chemische Wechselwirkungen zwischen UO_2 und Zircaloy-4 im Temperaturbereich von 1ooo bis 2ooo$^\circ$C. In KfK 347o, 1982.

Fig. 3: Separate Effects Tests on the Interactions between Zry Cladding and Oxide Fuel

Time Schedule



2.1.3 EQUILIBRIUM PHASE RELATIONS IN THE TERNARY U-ZR-O SYSTEM AT TEMPERATURES BETWEEN 1500 AND 2000 ^{o}C (A. Skokan, IMF I)

TEST OBJECTIVE

In addition to the separate effects tests on the interactions between Zry cladding and UO_2 described above, investigations to determine the U-Zr-O equilibrium phase relations in the temperature range between 1500 and 2000 ^{o}C are required for the interpretation of the UO_2/Zry chemical interaction experiment results. To date the isothermal sections at 1500 and 2000 ^{o}C and the UO_2-Zr(O) quasi-binary join are known, although there are still some discrepancies with respect to the solubility of UO_2 in the metallic melt. The test objective is to complete the available data and to eliminate the existing discrepancies.

EXPERIMENT CONDUCT

The experiments will be conducted on homogenized powder samples in high-temperature furnaces. The examination of the samples will be done by the following methods: optical microscope, X-ray diffraction, chemical analysis and microprobe analysis. Dynamic methods of investigation such as high-temperature differential thermal analysis and high-temperature X-ray diffraction will also be included.

STATUS OF WORK

The experimental work began in June 1982. Differential thermal analyses have been conducted on some samples along the UO_2-Zr(O) join in order to revise the eutectic temperature. These tests will be finished in early 1983.

2.2 OUT-OF-PILE SINGLE ROD AND BUNDLE EXPERIMENTS WITH INDIRECTLY HEATED FUEL ROD SIMULATORS (CORA PROGRAM) (S. Hagen, K. Hain, IT)

These tests are an extension of the 1976 - 1979 out-of-pile work of S. Hagen on the behavior of LWR fuel rod bundles at temperatures between LOCA and core melt down conditions.

OBJECTIVES

The objectives of the CORA Program are:

- to quantify the competing effects of cladding oxidation and formation of liquefied phases between Zry cladding and UO_2 fuel,
- to investigate the behavior of liquefied phases in fuel rod bundles and their interaction with steam,
- to determine the influence of spacers, absorber material and control rod guide tubes on the fuel element behavior,
- to investigate the fragmentation of severely embrittled and damaged fuel rods during quenching, and
- to perform out-of-pile-reference tests for in-pile experiments like PBF, PHEBUS and SUPER SARA.

RESULTS TO DATE

The 1976 - 1979 out-of-pile experiments by Dr. S. Hagen on the behavior of single rods and rod bundles at temperatures between LOCA and core melt down conditions have led to the following results:

- Fuel behavior beyond LOCA conditions is primarily influenced by interactions between the Zry cladding and other in-core materials, especially the formation of liquid phases and the dissolution of the UO_2 pellets by molten Zry beginning macroscopically at 1850 $^{\circ}C$.
- Steam oxidation of Zircaloy can produce a protective layer of ZrO_2 that can reduce the interaction of Zircaloy with other core materials, enclose a melt of the remaining Zircaloy, and if the cladding is fully oxidized, raise

the overall melt temperature of the cladding to between 2500 ^{o}C and 2650 ^{o}C. The extent of oxidation and embrittlement is strongly influenced by the steam supply and the initial heatup rate. The heatup rate is,in turn, influenced by the exothermic Zircaloy/steam reaction and possible heat losses through the core.

Recent 1981 - 1982 experiments by Dr. Hagen with single rods enclosed in Zircaloy shrouds and insulation have shown that temperature escalations to around 2200 ^{o}C are possible, but that inherent self-limiting mechanisms cause the temperature spike to turnaround. In addition to the formation of the oxide layer itself, such mechanisms may include runoff of the molten Zircaloy, steam starvation, and hydrogen blanketing.

- The early bundle experiments showed that the melt may refreeze between the rods as a solid mass. Experiments with absorber rods showed that the first movement of molten material was associated with the failure of the stainless steel absorber rods at 1400 ^{o}C. Again, the material refroze as a solid mass.

CORA CONCEPTUAL DESIGN

CORA is a new test facility designed to investigate the behavior of fuel bundles during severe fuel damage conditions.
The CORA design will include the capability of quenching the bundle and containing potential steam explosions. Table 1 shows the technical data of the CORA test facility. Figure 4 is a schematic diagram of the facility components and their interrelationships, while Figure 5 is a conceptual design of the main components within the containment. CORA will use internally heated electric fuel rod simulators of one meter heated length in a variety of configurations from single rods to 7 x 7, 37 rod bundles (see Fig. 6). In the bundle configuration a maximum of 24 heated rods is feasible. Heatup rates of 0,5 - 4 K/s are possible to a cladding maximum temperature of about 2000 ^{o}C. Ballooning and burst of the cladding will be possible plus operation with cracked fuel pellets. The bundle will be surrounded by a high temperature radiation shield to reduce radial heat losses. The atmosphere will normally be steam at atmospheric pressure with a possible maximum pressure of 12 bar.

Two important features of the design are the quenching and bundle inspection capabilities. Figure 7 shows the course of a complete experiment. After the heatup phase the bundle can be quenched by raising a quench funnel from below the bundle until a desired water level in the bundle is reached. When the experiment phase is complete, the radiation shield and the quench funnel can be lowered away from the bundle allowing the bundle to be inspected and photographed without moving it. For fragile or fragmented experiments this feature may prove very important.

EXPERIMENT MATRIX

Table 2 lists the current experiment matrix for CORA including some preliminary experiments to be conducted in Dr. Hagen's old facility, NIELS. The planned experiments shown should be considered as a guide to the future experiments rather than as a fixed program. CORA is a versatile facility and as the experiments are performed and evaluated it may be that the priorities for new tests will change. This is also true as the results of foreign experiments become available. Thus, the plan is intended to be flexible and represents current priorities and thinking.

TIME SCHEDULE

The preliminary time schedules for the construction of the CORA test facility and for the test performance are given in Fig. 8 and Fig. 9 .

REFERENCES FOR CHAPTER 2.2

/1/ S. Hagen, P. Hofmann, H. Malauschek, C. Politis, A. Skokan: Phenomena and material behavior during meltdown of PWR fuel rods. OECD/NEA Specialists Meeting on "The Behavior of LWR Fuel Elements under Accident Conditions", Spåtind, Norwegen (1976).

/2/ S. Hagen: Out-of-pile Experiments on the High Temperature Behaviour of Zry-4 Clad Fuel Rods. 6th Intern. Conference on Zirconium in the Nuclear Industry, 28 June-1 July 1982,Vancouver,British Columbia,Canada.

/3/ S. Hagen, unpublished report

/4/ K. Hain, F. Brüderle, W. Butzer, F. Schloß, T. Vollmer, unpublished report

Table 1:

Technical Data of the SFD - Experiment Facility CORA

Simulator diameter:	10,75 mm
Overall length:	2175 mm
Heated length:	1000 mm
Bundle size:	max. 37 rods
	7 x 7 arrangement
Pitch:	14,3 mm
Maximum number of heated rods:	24
Heatup rate:	0,5 - 4 K/s
Max. rod temperature:	2000 °C
Max. system pressure:	12 bar
Max. rod internal pressure:	100 bar
Single rod power:	16 kW, 160 W/cm
Bundle power:	96 kW
Average linear power:	40 W/cm
Steam supply:	0 - 12 g/s
Inlet steam temperature:	100 - 180 °C
Quench funnel lifting speed:	0 - 4 cm/s
Quench water temperature:	50 - 120 °C
Max. quench water amount:	70 ltr.

Table 2: TENTATIVE TESTMATRIX FOR OUT-OF-PILE SINGLE ROD EXPERIMENTS ON SEVERE CORE DAMAGE

Test No.	Heating Rate [°C/sec]	Max. Temp. [°C]	Atmosphere	Diff. Press. [bar]	State of the Pellets	Ballooning	End of Test	Facility	Objectives of Tests
1	0.2	2000	steam in excess	0	uncracked	No	slow cooldown inert gas	Niels	Influence of Heating Rate
2	0.5	"	"	"	"	"	"	"	
3	1.0	"	"	"	"	"	"	"	
4	2.0	"	"	"	"	"	"	"	
5	4.0	"	"	"	"	"	"	"	
6	0.5	"	steam starvation	"	"	"	"	"	Influence of Steam Starvation
7	2.0	"		"	"	"	"	"	
8	2.0	"	steam in excess	"	cracked	"	"	"	Influence of the State of Pellets
9	4.0	"	"	"	"	"	"	"	
10	2.0	"	"	10 o.p.	uncracked	"	"	CORA	Influence of Contact cladding/Pellet
11	4.0	"	"	"	"	"	"	"	
12	2.0	"	"	0	rubble	simulated ballooning	"	Niels	Influence of Ballooning on Dissolution
13	2.0	"	"	to be fixed	cracked	balloon. without burst	"	CORA	
14	2.0	"	"	90 i.p.	"	ball. at 800 °C (α-phase)	"	"	
15	2.0	"	"	10 i.p.	"	ball. at 1100 °C (β-phase)	"	"	

Table 2 (continued)

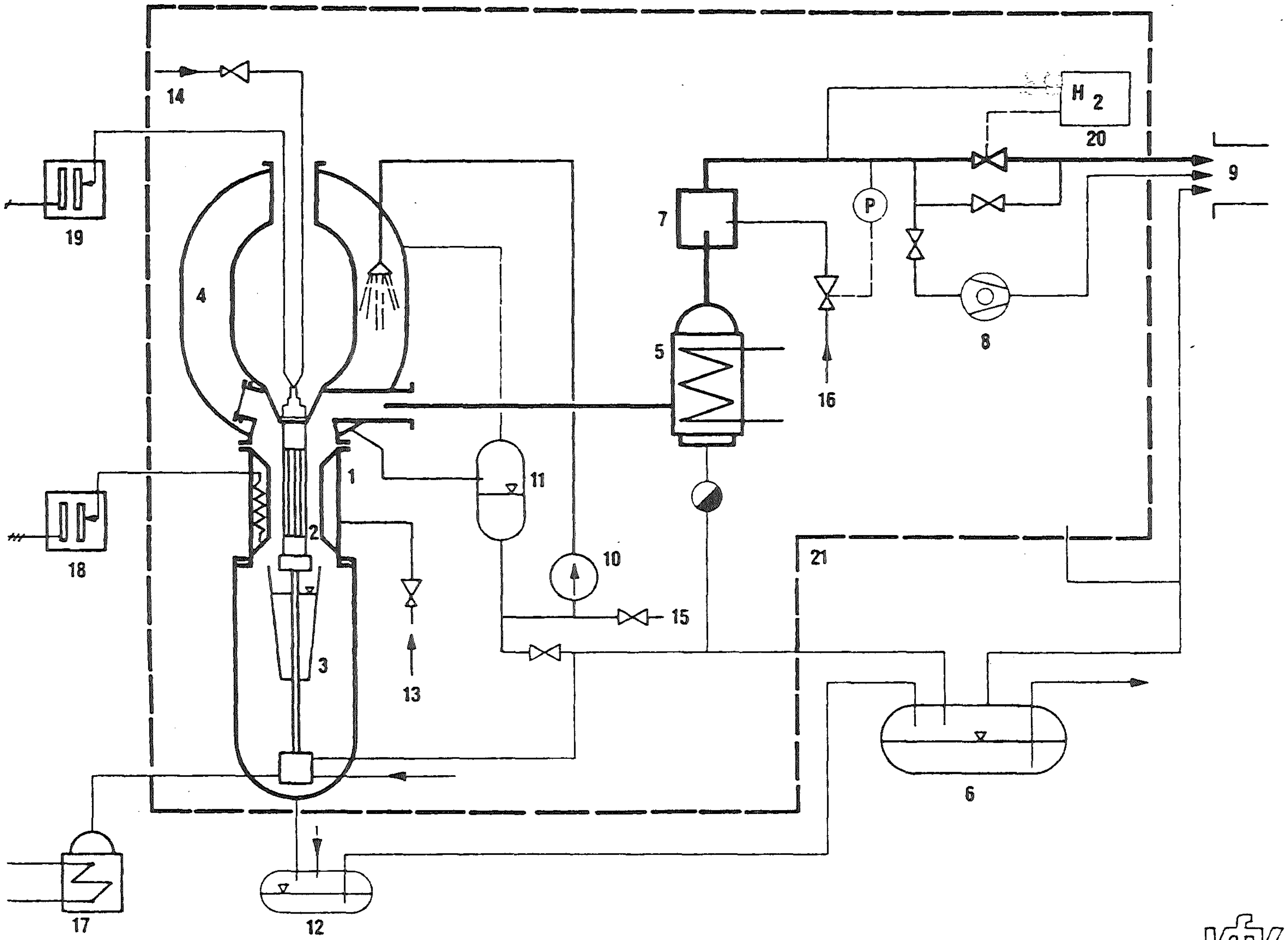
Test No.	Heating Rate [^{o}C/sec]	Max. Temp. [^{o}C]	Atmosphere	Diff. Press. [bar]	State of the Pellets	Ballooning	End of Test	Facility	Objectives of Tests
16	0.5	2000	steam in excess	0	uncracked	No	quench	CORA	Fragmentation by Quenching
17	4.0	"	"	"	"	"	"	"	
18	0.5	"	"	"	cracked	"	"	"	
19	4.0	"	"	"	"	"	"	"	
20	2.0	"	"	to be fixed	"	ballooning without burst	"	"	Influence of Ballooning on Fragmentation
21	2.0	"	"	90 i.p.	"	b. at 800 (α-phase)	"	"	
22	2.0	"	"	10 i.p.	"	b. at 1100 (β-phase)	"	"	

Table 2 (continued):

TENTATIVE TESTMATRIX FOR OUT-OF-PILE BUNDLE EXPERIMENTS ON SEVERE CORE DAMAGE

Test No.	Heating Rate [°C/sec]	Max.Temp. [°C]	Atmosphere	State of Pellet	Diff.Press. [bar]	Ballooning	End of Test	Facility	Objectives of Tests
101	0.5	2000	steam excess	cracked	0	No	slow	CORA	Influence of heating rate
102	0.5	"	"	"	"	"	quench	"	on dissolution and
103	4.0	"	"	"	"	"	slow	"	fragmentation.
104	4.0	"	"	"	"	"	quench	"	
105	0.5	"	"	"	90 i.p.	α-phase 800 °C	slow	"	
106	0.5	"	"	"	"	"	quench	"	Influence of ballooning
107	4.0	"	"	"	"	"	slow	"	on dissolution and
108	4.0	"	"	"	"	"	quench	"	fragmentation for different
109	0.5	"	"	"	10 i.p.	β-phase 1100 °C	slow	"	heating rates.
110	0.5	"	"	"	"	"	quench	"	
111	4.0	"	"	"	"	"	slow	"	
112	4.0	"	"	"	"	"	quench	"	

Test No.	Heating Rate [°C/sec]	Max.Temp. [°C]	Atmosphere	State of Pellet	Guide Tube	Absorber	End of Test	Facility	Objectives of Test
113	2.0	1750	steam excess	Al_2O_3	stainless steel	Ag/In/Cd	slow	CORA	Influence of absor ber rods
114	2.0	"	"	"	"	"	quench	"	and guide tubes on
115	2.0	"	"	"	"	Boron Silicate Glas	slow	"	blockage and fragmentation
116	2.0	"	"	"	"	Grey Absorber	slow	"	
117	2.0	"	"	"	"	Empty Tube	slow	"	
118	2.0	"	"	"	Zry	Ag/In/Cd	slow	"	
119	2.0	"	"	"	"	"	quench	"	
120	2.0	"	"	"	"	Boron Silicate Glas	slow	"	



1 high temperature shield
2 rod bundle
3 quench facility
4 surge condenser
5 vent condenser
6 condensat container
7 mixing chamber
8 vacuum pump
9 exhaust gas system
10 spray pump
11 spray collector
12 condensat transfer tank
13 Ar-input
14 fisson gas simulation
15 water supply
16 inert gas
17 steam generator
18 HT-shield power supply
19 bundle power supply
20 H 2 -detection
21 containment

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FIG. 4: **SFD - experiment facility CORA - schematic**

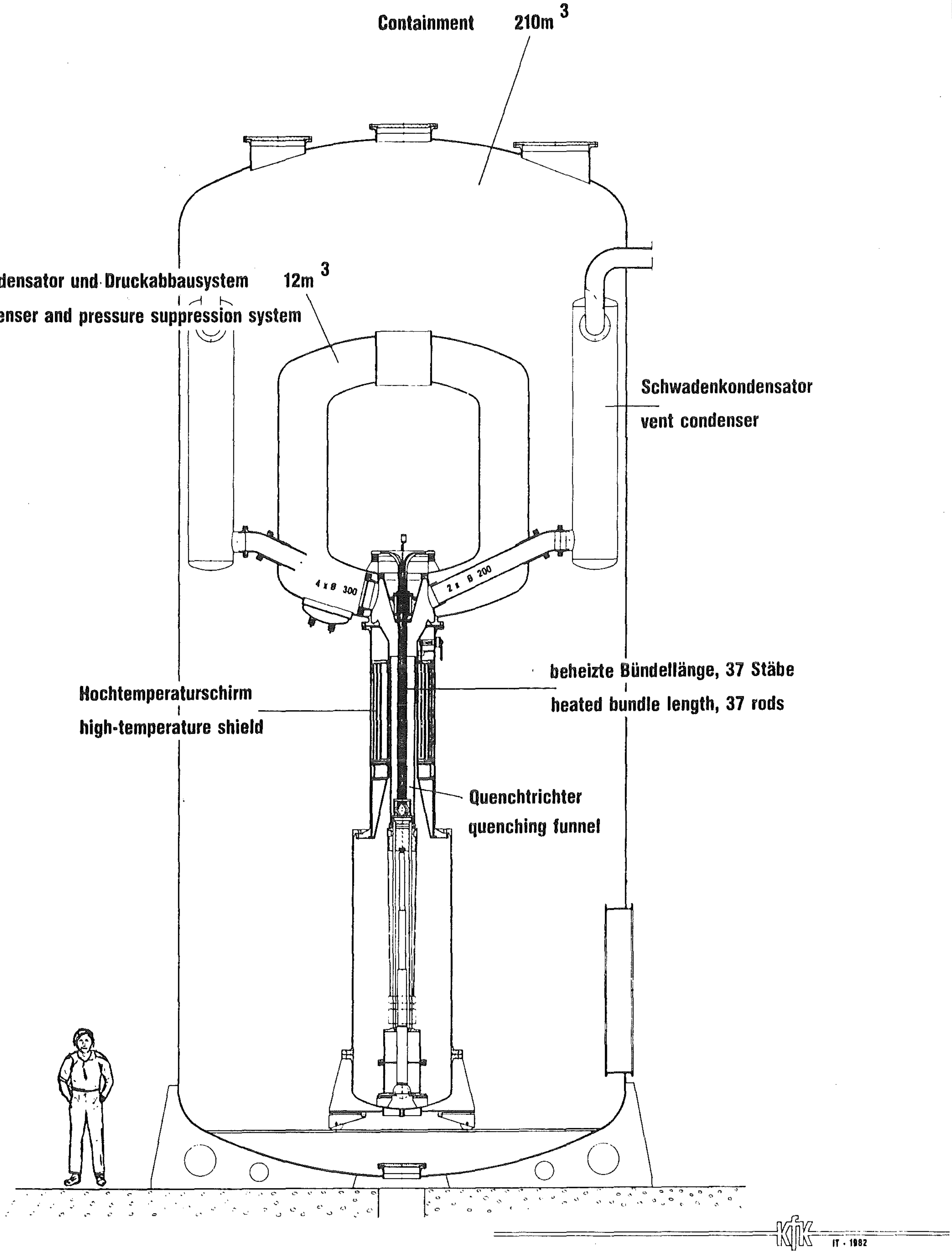

FIG. 5 Severe fuel damage - Versuchsanlage CORA

PROJEKT NUKLEARE SICHERHEIT PNS

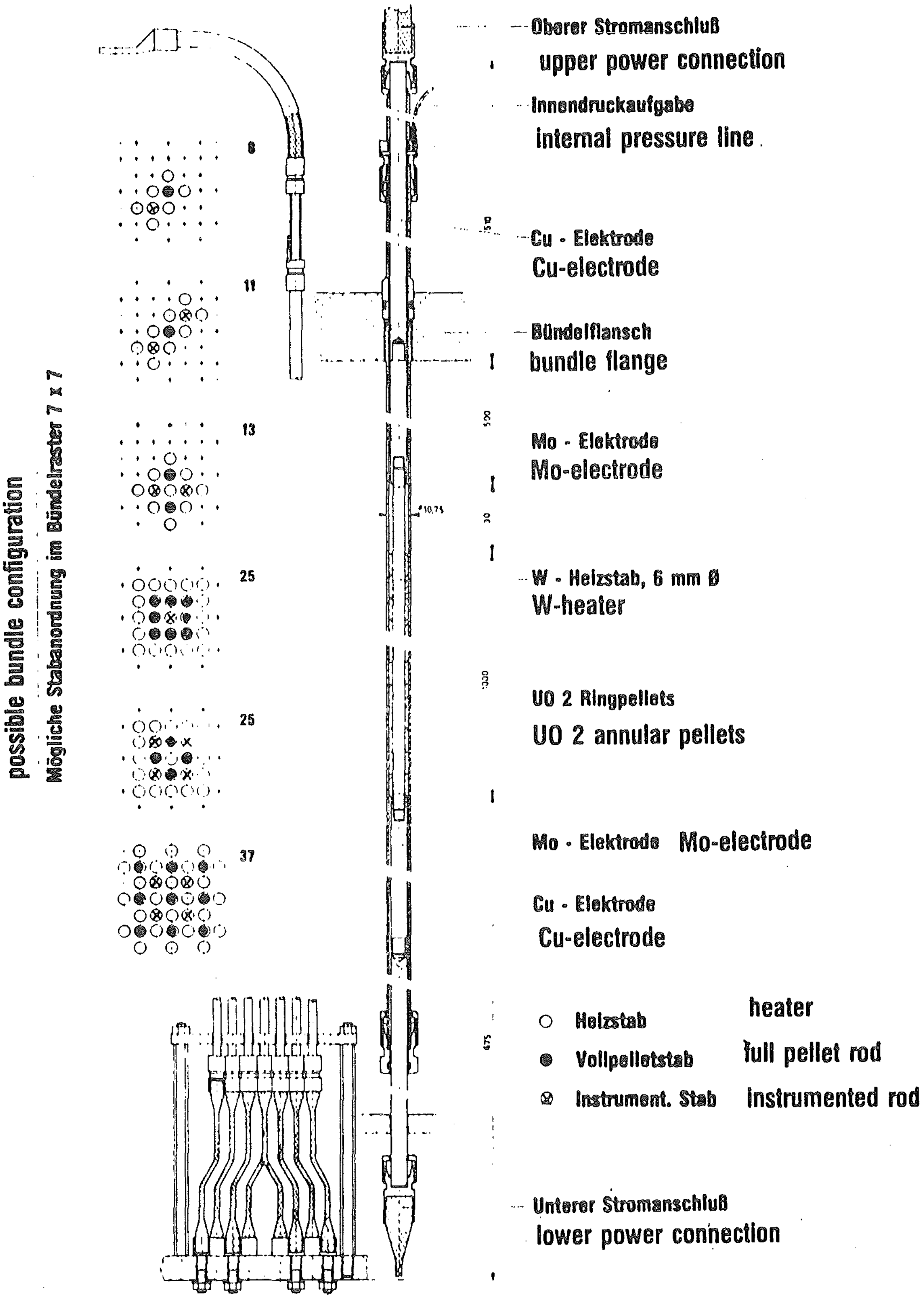

FIG. 6:

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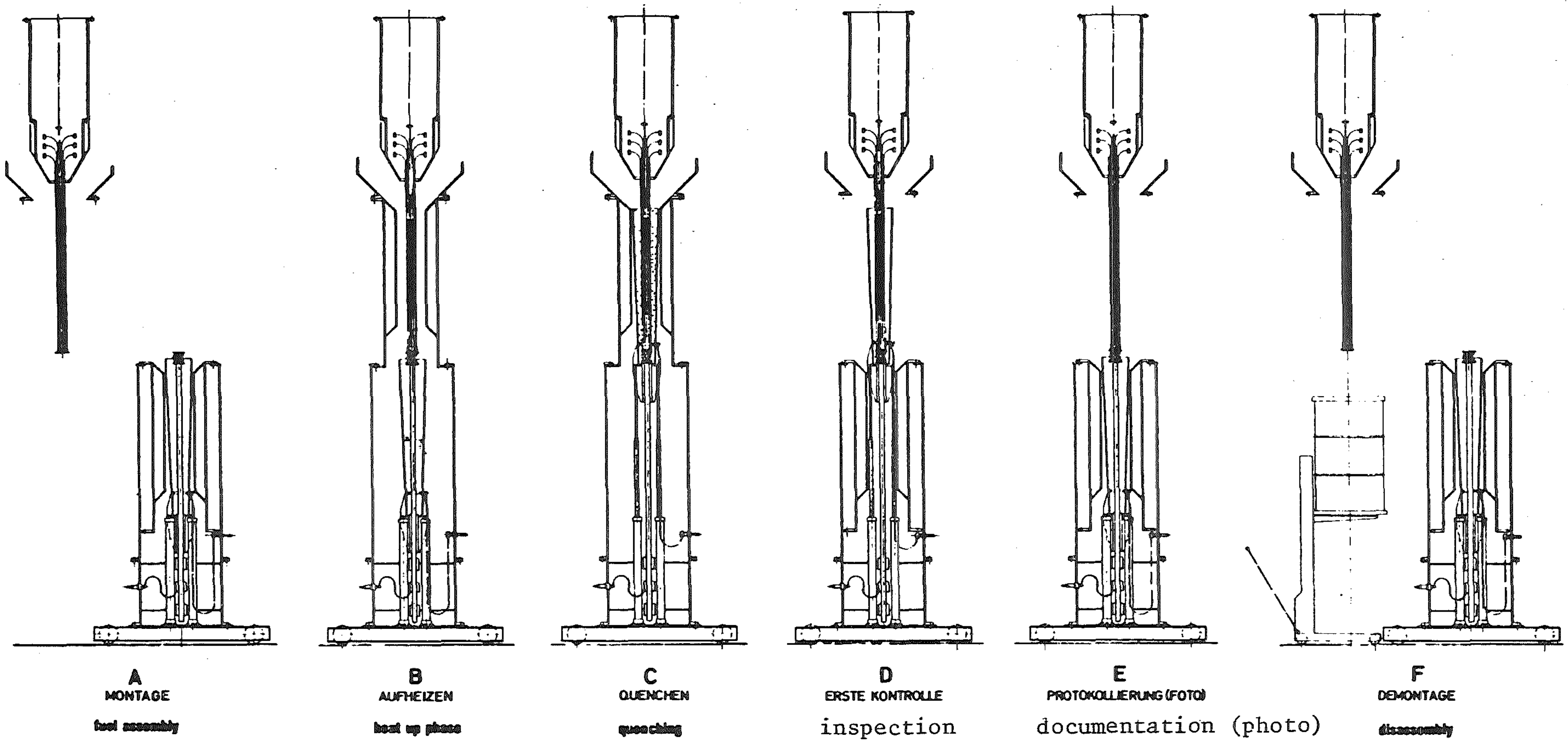
SFD - Versuchsanlage CORA
Beheizter Brennstabsimulator
SFD - experiment facility - heated fuel rod simulator



KfK IT - 1981

FIG. 7: SFD - VERSUCHSANLAGE CORA
VERSUCHSABLAUFSCHRITTE

SFD - TEST FACILITY CORA
TEST PROCEDURE

Fig. 8: Time Schedule for the Construction of the SFD Experiment Facility CORA

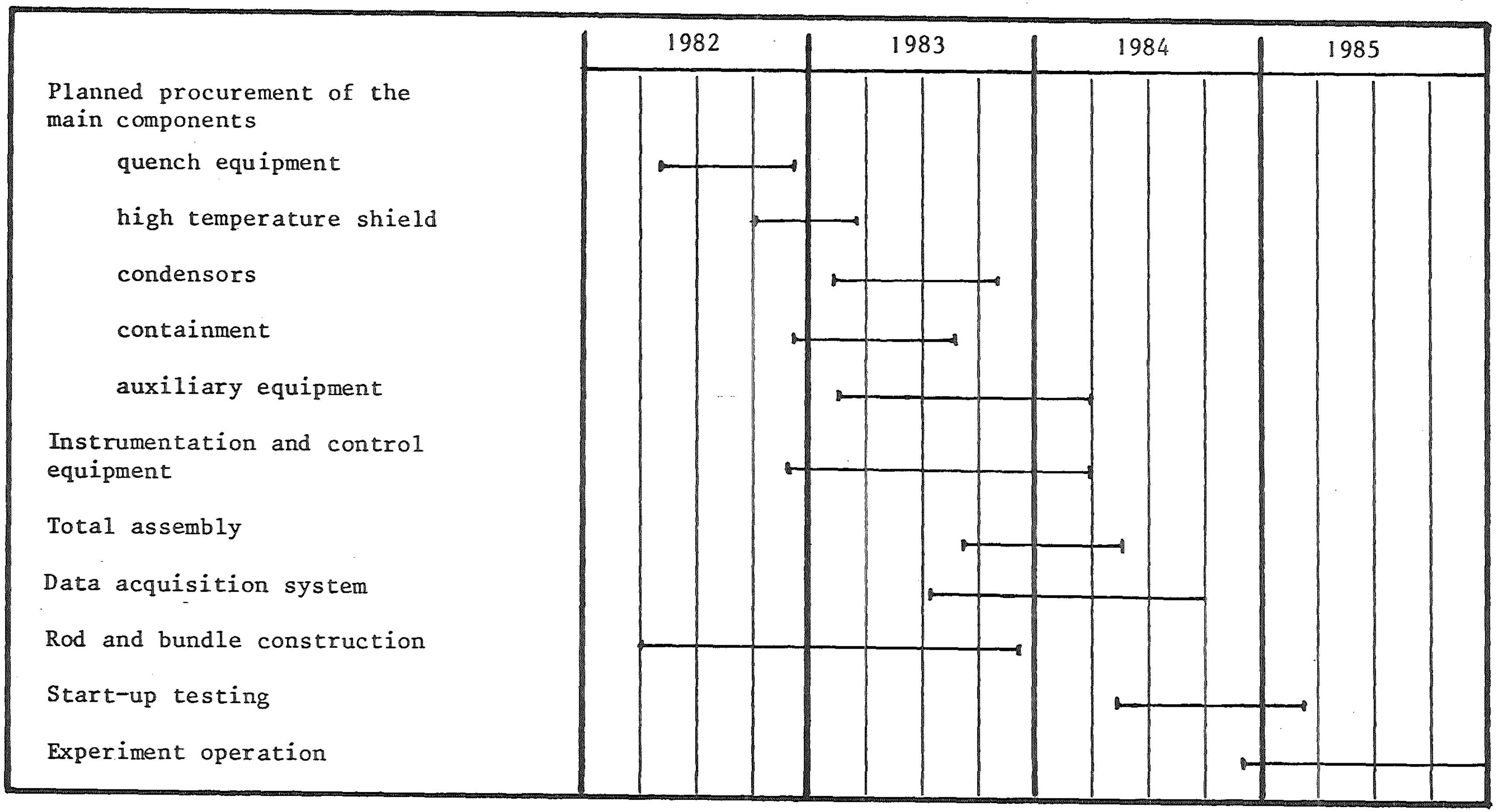


Fig. 9:

Time Schedule for Out-of-Pile Severe Fuel Damage Tests (CORA-Program)

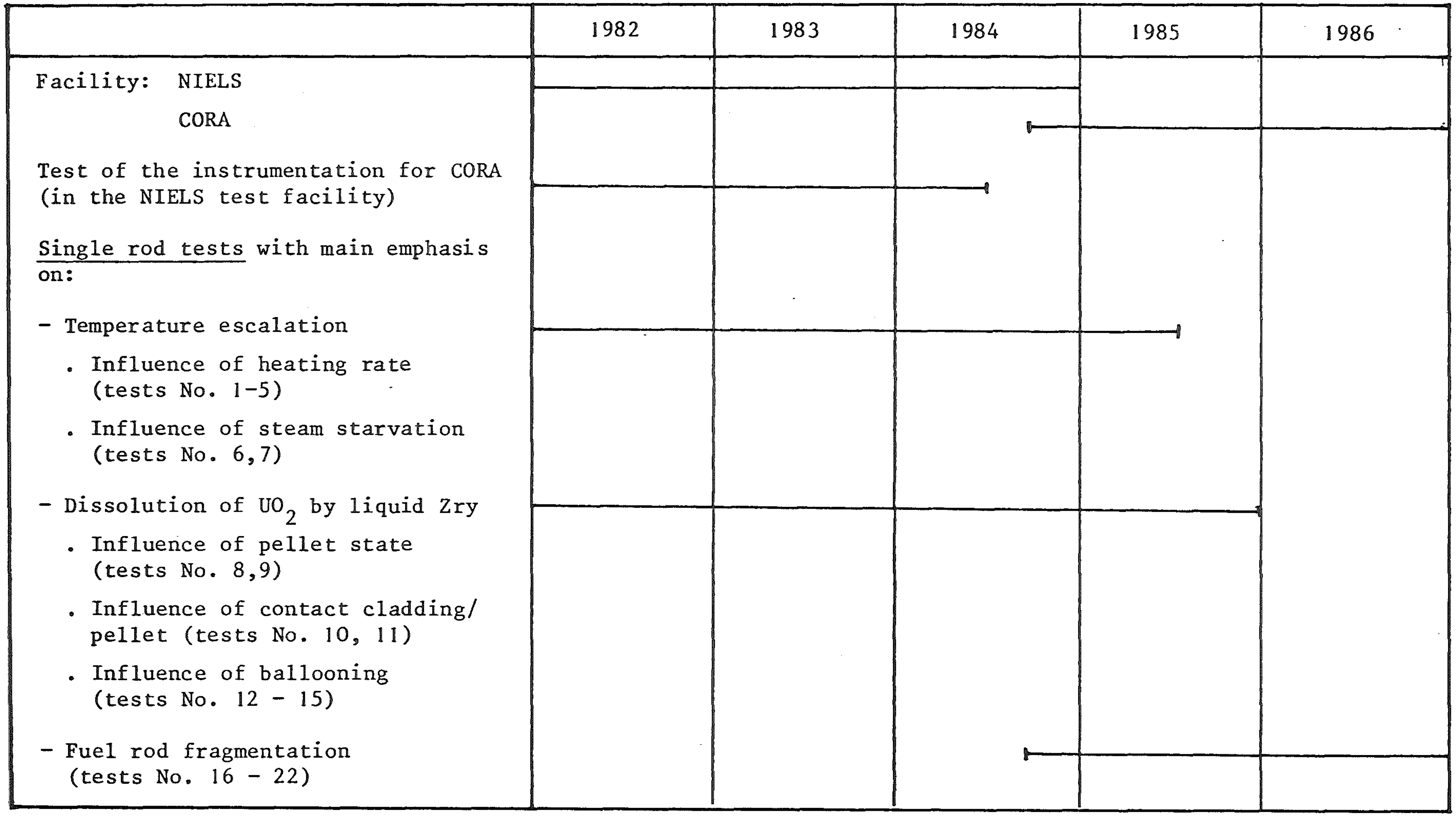
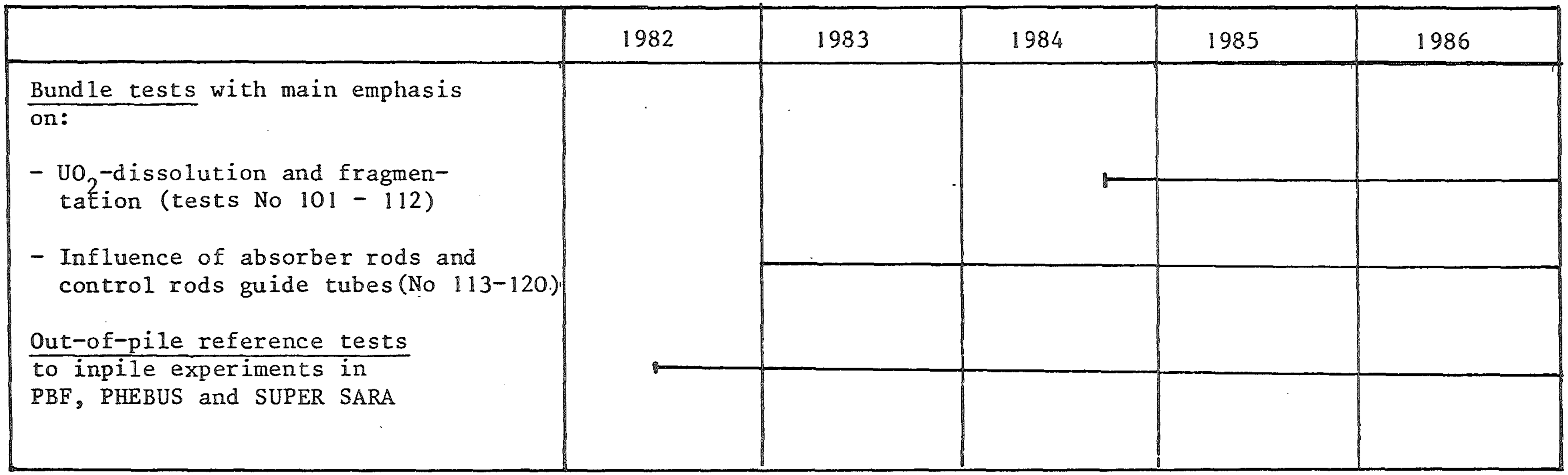


Fig. 9 (continued):

Time Schedule for Out-of-Pile Severe Fuel Damage Tests (CORA-Program)



2.3 OUT-OF-PILE TESTS ON THE LONG TERM COOLABILITY OF SEVERELY DAMAGED CORE GEOMETRIES (COLD PROGRAM , G. Hofmann, IRB)

PROGRAM OBJECTIVES

- Investigation of the dryout mechanisms in very deep particulate beds to improve the understanding of the physical phenomena involved.

- Determination of the dryout heat fluxes for deep beds with different hydraulic boundary conditions; combinations of the liquid replenishment through the top and the bottom of the bed (top-fed and bottom-fed, respectively).

- Model improvement and verification on the basis of the experimental results.

PROGRAM PLAN

1. Scoping experiments with a 0,5m deep bed under natural recirculation bottom-fed conditions.

2. Experimental investigation of the influence of bed depth on the dryout heat flux in up to 0,5m deep top-fed beds.

3. Experimental investigation and modelling of the location of incipient dryout in top-fed beds of 3mm particles.

4. Dryout experiments in up to 1m deep beds.

5. Dryout experiments with gradually increased liquid flow rates from below (combined top/bottom-feeding).

6. Determination of the axial pressure and saturation profiles within deep beds.

7. Comparison of experimental results with existing models and model improvement where necessary and possible.

EXPERIMENTS

Experiments are performed with beds of inductively heated stainless steel particles of uniform size and water as coolant; particle sizes which are currently in the program are 3mm and one smaller diameter on the order of 1 to 2mm.

Irregularly broken particles might be included in the future. Bed diameter is about 6cm and bed heights are expected to range up to 1m.

Bottom inlet flow is by gravity feed. Temperature and pressure measurements are done with thermocouples and transducers, respectively; it is aimed at the measurement of the saturation with an appropriate technique.

The critical dryout heat flux is determined with a series of experiments as the highest heat flux for which the whole bed stays at saturation temperature under steady state conditions. For power steps of different size beyond critical the transient behavior is recorded from the measurement divices and from visual observation.

Post-dryout measurements are limited to a maximum temperature of about 500°C.

STATUS AND TEST SCHEDULE AS OF AUGUST 1982

Experiments on items 1 through 3 listed above are finished and the main results are reported.

Work is in progress to increase the capability of the induction generator to heat up to 1m deep beds. The present time schedule for the continuation of the program is given on the attached table (Fig. 10).

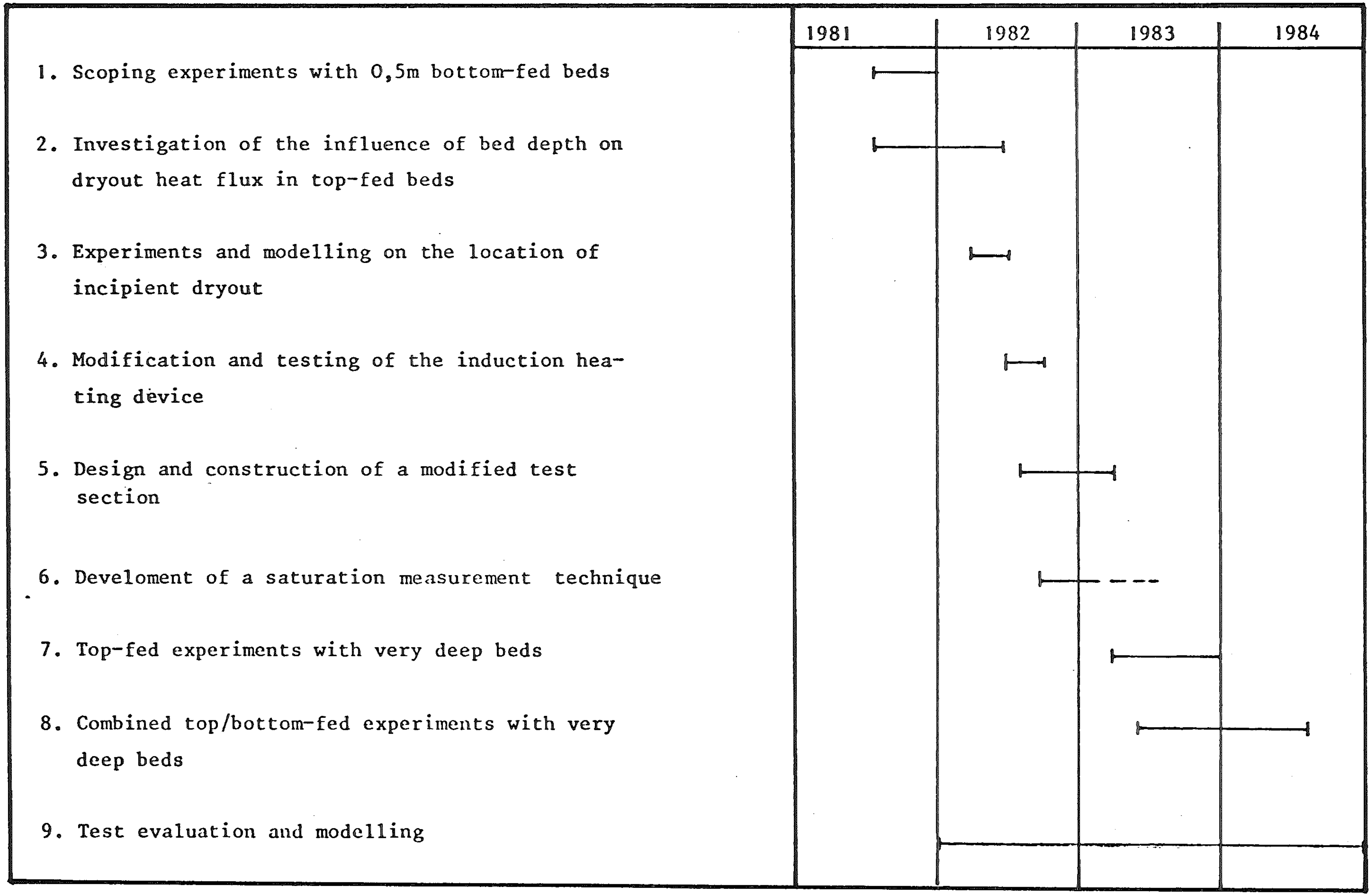
REFERENCES FOR CHAPTER 2.3

/1/ G. Hofmann: On the Location and Mechanisms of Dryout in Top-Fed and Bottom-Fed Particulate Beds. Fifth Post Accident Heat Removal Information Exchange Meeting, 28 - 3o. Juli 1982, Karlsruhe.

Fig. 10:

OUT-OF-PILE TESTS ON THE LONG TERM COOLABILITY OF SEVERELY DAMAGED CORE GEOMETRIES (COLD PROGRAM)

TIME SCHEDULE (AUGUST 1982)



2.4 ANNEALING EXPERIMENTS ON FISSION GAS RELEASE AT HIGH TEMPERATURES

(H. Zimmermann, IMF I)

TEST OBJECTIVE

Investigation of the fission gas and fuel behavior at LOCA and SFD typical temperatures. Fission gas release and swelling are investigated as functions of the temperature, of the time the fuel is at this temperature, and of the burnup of the fuel.

TEST PROGRAM

Test material is the UO_2 of the fuel rods from the FR 2 LOCA-Tests (PNS project 06.01.08). The test program includes three parts:

- Characterization of the fuel in the irradiated state (0.3, 0.9, 2.4, 3.7, and 3.9 % burnup).
- Examination of the fuel in the LOCA-tested rods.
- Annealing tests with irradiated fuel samples taken from non-LOCA-tested fuel rods. The aim of the annealing tests is to simulate small break LOCA conditions. The samples are annealed in Mo or Nb capsules for various times at temperatures between 1200 and 2000 ^oC.

Examination methods are:

- Density measurements to evaluate the swelling of the fuel.
- Determination of the released fission gas.
- Determination of the retained fission gas in two steps by measuring the amount of fission gas released (1) during grinding the fuel ("gas in pores") and (2) during dissolution of the powdered fuel ("gas in the matrix") in order to investigate the redistribution of fission gas during the annealing.
- Examination of the microstructure.

STATUS OF THE WORK AND TEST SCHEDULE

The characterization of the fuel in the irradiated state and the examination of the LOCA-tested fuel rods have been completed. The annealing tests are in progress. Evaluation of the first test results of the annealing tests at temperatures up to 1600 °C has begun. The present schedule for the tests is given in the following table (Fig. 11).

	1982	1983	1984	1985
Annealing tests at temperatures up to 1600 °C				
Annealing tests at temperatures up to 2000 °C (inductive heating)				
Evaluation of the test results				

2.5 SFD MODELING, CODE DEVELOPMENT AND ANALYSIS (W. Gulden, PNS-PL)

OBJECTIVES

To analyse the behavior of fuel rods and fuel rod simulators under SFD conditions in the frame of the NIELS and CORA facilities some new models have to be added to the code system SSYST-4. For this purpose already existing relevant computer code models will be integrated into SSYST (e.g. from EXMEL and SCDAP). Some other models mainly concerning the high temperature oxidation and the heat and mass tranfer between the oxidizing cladding and slowly flowing steam have to be developed.

The verification of these computer models will be based on the separate effects tests for high temperature oxidation of Zry and for interaction between Zry and UO_2, and on the experiments of the NIELS and CORA facility.

Pre- and posttest calculations will support the KfK-SFD experiments.

PROGRAM PLAN

The modelling, code development and analysis effort covers the following areas:

1. Integration of the relevant EXMEL modules into SSYST-4

   Modeling of SFD (based on old data reported by Hagen) has been previously conducted at the University of Stuttgart. An extended version of the code system SSYST was used to evaluate Hagen's experiments. The modules for rod melting, rod failure and candling produced at the University of Stuttgart are called EXMEL.

   As a first step of the KfK-SFD development some relevant modules of EXMEL will be integrated into SSYST-4.

2. Model development SSYST-4 single rod

2.1 High temperature oxidation kinetics (MULTRAN)
This module is based on the SIMTRAN code developed by S. Malang.

It calculates twoside cladding oxidation kinetics including:

- multilayer oxidation
- diffusion in ZrO_2, α-Zr, and β-Zr
- oxidation profile in the cladding
- influence of limited steam supply on the oxidation profile

2.2 Formation and behavior of liquified phases between Zry and UO_2.

This module is an extension of the module MULTRAN to take into account the fuel cladding interaction. It will be based on the experiments of P.Hofmann on reaction kinetics between UO_2 and Zry-4. In addition to the features of MULTRAN it will contain models for:

- reaction kinetics between UO_2 and Zry
- dissolution of UO_2 by molten Zr
- formation of liquified rod areas

2.3 Heat and mass transfer between oxidizing cladding and steam.

This model will look in detail on the concentration profile of hydrogen and steam in the channel near the cladding surface. As typical for SFD conditions, steam velocity is assumed to be low.

The module will contain models for:

- hydrogen and steam concentration profiles in the channel
- steam starvation
- hydrogen blanketing

2.4 Modelling of special NIELS and CORA features.

This acitivity concerns the modelling of special phenomena and geometrical features of the NIELS and CORA facilities.

- power generation; heat production in the electrical heater rod as a function of temperature distribution in the heater rod
- radial temperature distribution; modelling of shroud, insulation and flowing or stagnant steam between cladding, shroud and insulation.

3. Pre- and posttest calculations for NIELS and CORA.

These activities cover the following areas:

- verification of the SFD computer codes
- assistance in the design of the CORA facility
- assistance in understanding the different SFD-phenomena in NIELS and CORA

4. Tentative activities

Due to the present plan new models on solidification of the melt and stop-and-go melt-freeze during candling will be developed only if related experiments show reproducible results and demonstrate a need. The same holds for models concerning the loss of fuel rod geometry due to brittle failure and meltdown-processes.

The development of models for SFD bundle behavior is not yet decided. A possible alternative could be the use of the SCDAP-code, being developed at EG&G, Idaho.

STATUS OF THE WORK AND TIME SCHEDULE

The code system SSYST was developed within the PNS project for fuel rod analysis under accident conditions. For the versions available (SSYST-2 and SSYST-3) main emphasis was put on analysing the behavior of fuel rods and fuel rod simulators under LOCA conditions and under conditions in experimental facilities for LOCA-relevant investigations (e.g. REBEKA, COSIMA, FR2-inpile, PBF-LOC-test).

The existing SSYST-models, however, are still valid for other transient conditions than LOCA as long as high temperature oxidation is not the dominating phenomenon.

Version SSYST-4 being developed at the moment is dedicated to fuel rod analysis under SFD conditions. It will use the existing SSYST-models up to approx. 12oo ^{o}C. Models for detailed analysis of the oxidation process at high temperatures, the fuel cladding interaction and the melting and embrittlement process are being developed. This development is done in close connection to the experiments in the NIELS and CORA facilities, to the separate effects tests on high temperature oxidation of Zry and on the interaction between Zry cladding and oxide fuel.

A first version of the high temperature oxidation module MULTRAN is available.

The present schedule of the individual tasks is given in the attached table (Fig. 12).

Fig. 12:

Time Schedule for Code Development and Analysis